Technical Program for Long-Term Management of Canada's Used Nuclear Fuel – Annual Report 2010

NWMO TR-2011-02 March 2011

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Nuclear Waste Management Organization

NUCLEAR WASTE MANAGEMENT ORGANIZATION SOCIÉTÉ DE GESTION DES DÉCHETS NUCLÉAIRES

Nuclear Waste Management Organization
22 St. Clair Avenue East, 6th Floor
Toronto, Ontario
M4T 2S3
Canada

Tel: 416-934-9814
Web: www.nwmo.ca

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ABSTRACT

Title: **Technical Program for Long-Term Management of Canada's Used Nuclear Fuel – Annual Report 2010**
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Author(s): J. McKelvie, K. Birch, J. Freire-Canosa, M. Garamszeghy, F. Garisto, P. Gierszewski, M. Gobien, S. Hirschorn, N. Hunt, M. Jensen, A. Khan, E. Kremer, G. Kwong, T. Lam, L. Lang, C. Medri, A. Murchison, S. Russell, U. Stahmer, E. Sykes, J. Villagran, A. Vorauer, T. Wanne and T. Yang
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Abstract

This report is a summary of progress in 2010 for the Nuclear Waste Management Organization's (NWMO's) Technical Program. The Technical Program is supporting implementation of Adaptive Phased Management (APM), Canada's approach for long-term management of used nuclear fuel.

Significant technical program achievements in 2010 include:

- The NWMO Independent Technical Review Group (ITRG) held their third annual review of the NWMO technical program. The ITRG report noted significant development in the NWMO's technical program since 2008 and indicated that the program covers a full range of scientific and technical topics that are relevant to the current stage of APM implementation. NWMO prepared an action plan addressing the recommendations of the ITRG report. The ITRG 2010 report and NWMO action plan are available on the NWMO website.

- NWMO continued to participate in international research activities associated with the SKB Äspö Hard Rock Laboratory, Mont Terri Rock Laboratory, Greenland Analogue Project, Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency Research Projects and the international working group on biosphere modelling (BIOPROTA).

- NWMO provided research contracts and research grants to 13 Canadian universities and, as an approved industrial partner with the Natural Science and Engineering Research Council of Canada, supported 6 postgraduate scholarships in 2010.

- NWMO's research program published 22 NWMO technical reports and 14 peer-reviewed journal articles.

- NWMO conducted research on: used fuel container corrosion; repository sealing material development and repository design. NWMO also continued to develop a repository monitoring and retrieval program and continued to survey developments in used fuel reprocessing and alternative waste management technologies.

- NWMO continued to refine engineering conceptual designs, cost estimates, transportation logistics and implementation schedules in support of APM.

- In May 2010, NWMO issued "*Moving Forward Together: Process for Selecting a Site for Canada's Deep Geological Repository for Used Nuclear Fuel*" which describes the site selection process and the proposed site evaluation criteria to ensure safety of people and the environment.

- The NWMO geoscience program continued to develop plans and methods for detailed site investigations in the fields of: geochemistry; radionuclide transport properties; microbiology; geomechanics; seismicity; and hydrogeology. NWMO also continued to develop numerical modelling methods and continued to assess long-term geosphere stability associated with glaciation, seismicity and deep groundwater flow systems.

- NWMO continued to maintain and improve models and data suitable for supporting the safety assessment of potential sites and designs. In 2010, safety assessment models assessing the used nuclear fuel waste form, deep geological repository, generic geosphere and surface biosphere were improved, integrated and maintained.

- In 2010, the glaciation scenario postclosure safety case was published as two NWMO technical reports (TR 2010-09 and TR-2010-10). The study indicates that potential releases from a deep geological repository would remain well below regulatory limits even when the glaciation effects are considered. In addition, work on preparing the "Fourth Case Study" assessing the postclosure safety of a deep geological repository at a hypothetical site in crystalline rock continued and two preclosure studies were initiated to examine aspects of conventional and public radiological safety at the facility.

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1. INTRODUCTION

The Nuclear Waste Management Organization (NWMO) is implementing Adaptive Phased Management (APM), the approach selected by the Government of Canada in 2007 for long-term management of used nuclear fuel (NWMO, 2005). The APM Technical Program is focussed on developing preliminary designs, safety cases, cost estimates and associated research activities for a used fuel deep geological repository in order to ensure continuous improvement and consistency with best practices.

Examples of conceptual designs for a deep geological repository are illustrated in Figure 1.1 and Figure 1.2.

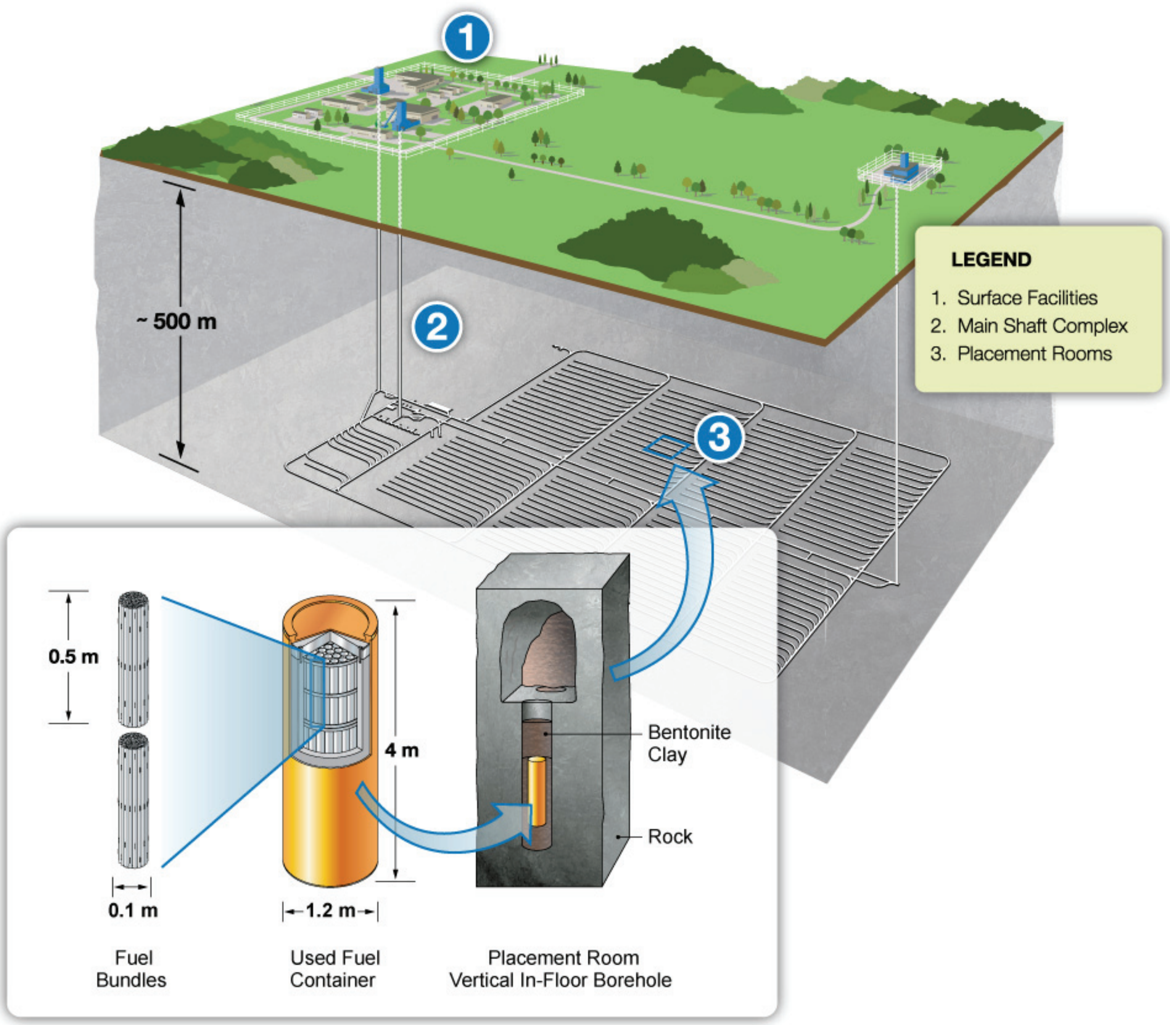


Figure 1.1: Illustration of a deep geological repository – In-floor borehole placement

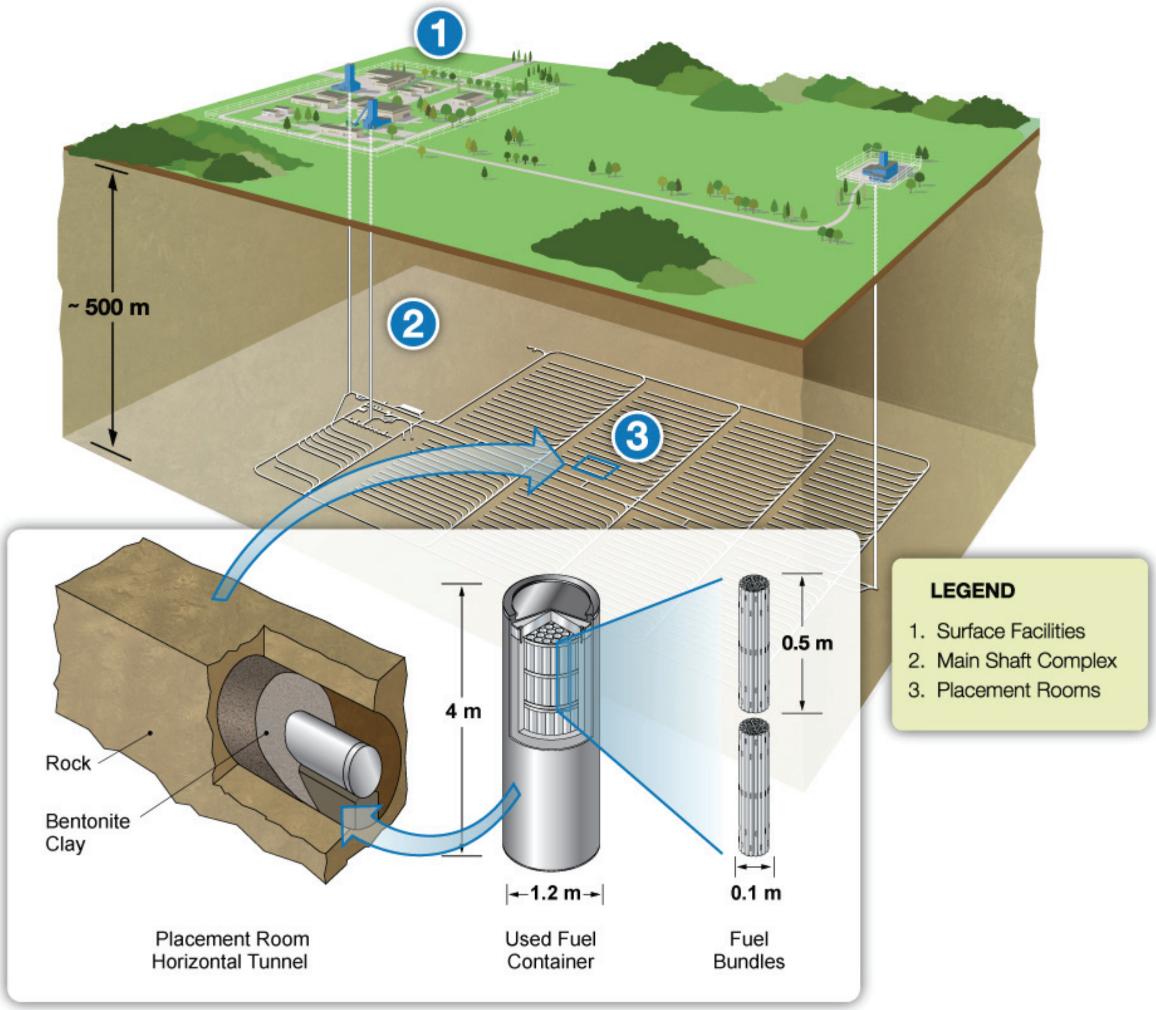


Figure 1.2: Illustration of a deep geological repository – Horizontal tunnel placement

This report summarizes progress in 2010 for the APM Technical Program. The APM Technical Program is described in further detail in the NWMO report "RD&D Program 2011 – NWMO's Program for Research, Development and Demonstration for Long-Term Management of Used Nuclear Fuel" (Villagran et al., 2011).

2. OVERVIEW OF CANADIAN RESEARCH AND DEVELOPMENT PROGRAM

2.1 REGULATORY FRAMEWORK

Nuclear facilities, including those for long-term waste management such as a deep geological repository (DGR), are regulated by the Canadian Nuclear Safety Commission (CNSC), under the *Nuclear Safety and Control Act (NSCA)*. Pursuant to regulations under this *Act*, licences are required from the CNSC for all phases of a project - site preparation, construction, operation, decommissioning, and abandonment. The CNSC provides additional guidance through regulatory policies, standards and guides.

A facility for long-term management of used fuel is subject to all of the requirements of nuclear safety and security and safeguards embodied in the *NSCA* and its associated Regulations. Also applicable is the CNSC's Regulatory Policy P-290, *Managing Radioactive Waste,* which states the following principles:

- Minimisation of waste generation;
- Management commensurate with the hazard;
- Assessment of future impacts to encompass the time of maximum predicted impact;
- Predicted impacts no greater than the impacts that are permissible in Canada at the time of the regulatory decision;
- Measures for safe management to be developed, funded and implemented as soon as reasonably practicable; and
- Trans-border effects no greater than the effects experienced in Canada.

CNSC's Regulatory Guide G-320, "*Assessing the Long Term Safety of Radioactive Waste Management",* describes approaches for assessing the potential impact that long-term radioactive waste management methods may have on the environment and on the health and safety of people.

The application for a CNSC licence for a DGR for used nuclear fuel would trigger an Environmental Assessment (EA) under the *Canadian Environmental Assessment Act (CEAA)*. Under the CEAA, an EA is required to assess the environmental effects of most projects requiring federal action or decisions. Public input will be required at appropriate stages in the EA and licensing process.

For the pre-project phase, the CNSC established a special project arrangement with the NWMO in March 2009, which includes CNSC review of NWMO information on conceptual APM design to identify any regulatory concerns. CNSC staff have agreed to review APM conceptual designs and illustrative safety assessments in representative host rock formations, one in crystalline rock and one in sedimentary rock. The NWMO will subject the submission to an independent peer review prior to providing it to CNSC. The NWMO submitted a report outline to the CNSC in November 2010 that will be used to document how the elements of G-320 have been incorporated into the development phase of the APM conceptual design and illustrative postclosure safety assessment. The objective of the pre-project review is to obtain a CNSC statement on their view of repository design and safety of the conceptual design in both potential host rock formations by 2013.

2.2 TECHNICAL PROGRAM OBJECTIVES & OVERVIEW

The primary objective of the APM Technical Program is to complete the preliminary designs, safety cases, cost estimates and generic research activities for a used fuel deep geological repository to support a licence application following planned selection of a preferred site in 2018.

To support the primary objective of the APM Technical Program, the following Technical Program objectives have been developed:

(1) Prepare updated generic reference designs, cost estimates and safety cases for a deep geological repository in crystalline rock and in sedimentary rock;

(2) Further improve the reference designs for a deep geological repository in crystalline rock and in sedimentary rock;

(3) Further increase confidence in the deep geological repository safety case;

(4) Obtain Canadian Nuclear Safety Commission pre-project review of reference designs and safety cases for a deep geological repository in crystalline rock and in sedimentary rock by 2013;

(5) Enhance scientific understanding of processes that may influence repository safety;

(6) Evaluate the adequacy of potential candidate sites for a deep geological repository by conducting site characterizations and safety evaluations; and

(7) Maintain awareness of advances in technology development and alternative methods for long-term management of used nuclear fuel.

The two main work areas supporting these objectives are:

a) Deep geological repository design development and the safety case, which will ensure NWMO has the appropriate level of engineering design and safety to support a future application of a site licence (objectives 1, 2 and 4); and

b) Building confidence in the safety case and improving our understanding of processes in the repository that may affect safety in the near- and long-term (objectives 3 and 5).

Important technical work is also planned in the area of geoscientific site characterization and evaluation to support site screening, feasibility studies of candidate sites and planned selection of a preferred site by 2018. As well, work will be conducted to maintain awareness of alternative technologies and new developments for managing used fuel over the long-term.

The repository engineering design, geoscience and repository safety technical program activities are described in more detail in Sections 3, 4 and 5, respectively.

A list of the technical reports produced by NWMO in 2010 is provided in Appendix A.1. Their respective abstracts are provided in Appendix B. All technical reports published before 2000 are listed in Garisto (2000), while the 2000 to 2009 reports are listed in corresponding annual progress reports (Gierszewski et al., 2001, 2002, 2003, 2004a; Hobbs et al., 2005, 2006;

Russell et al., 2007; Birch et al., 2008, Kremer et al., 2009, McKelvie et al., 2010). Note that prior to 2007, the technical program was managed by Atomic Energy of Canada Limited and then by Ontario Power Generation.

Appendix A.2 provides a list of the publications and presentations made by technical program staff and contractors. Appendix A.3 provides a list of graduate students awarded industrial postgraduate scholarships. Appendix A.4 provides a list of the primary external contractors and collaborators for the technical work program.

2.3 SUMMARY OF INTERNATIONAL ACTIVITIES

An important part of the NWMO's technical program is interacting with the corresponding national radioactive waste management organizations in other countries. The NWMO has formal agreements with SKB (Sweden), POSIVA (Finland), NAGRA (Switzerland) and ANDRA (France) to exchange information arising from their respective programs on nuclear waste management. These countries are developing used fuel repository concepts that are similar to the Canadian concept, and their programs are advanced with respect to repository siting, design development and regulatory approvals.

Since 2004, Canada has been participating in experiments associated with the SKB Äspö Hard Rock Laboratory. The purpose of this participation is to improve our understanding of key processes in a repository in crystalline rock through involvement in large-scale projects. This facilitates collaboration and sharing of lessons learned in repository technology development and site characterization. Specifically, in 2010, NWMO participated in the:

- Bentonite Colloid Transport Project;
- LASGIT Gas Injection Test;
- Engineered Barrier System Modelling Task Force; and
- Groundwater Modelling Task Force.

NWMO is a partner in the Mont Terri Project, which involves of a series of experiments carried out in the Mont Terri rock laboratory in Switzerland. The main aims of the project are: to test and improve techniques for hydrogeological, geochemical and geotechnical investigations in an argillaceous formation; to characterize the Opalinus Clay and its behaviour; to explore the interactions between the Opalinus Clay and other materials (such as engineered barriers); and to carry out research relevant to repository implementation (e.g., demonstration experiments). The experiments being conducted at Mont Terri have relevance for NWMO site characterization, engineering and safety assessment activities. Involvement in the Mont Terri Project allows NWMO to participate in state-of-science research in collaboration with 14 other international project partners, which includes several waste management agencies. NWMO is currently involved in the following experiments within the Mont Terri Project:

- Disturbances, Diffusion and Retention (DR-A);
- Determination of Stresses (DS);
- Gas Path Through Host Rock and Along Seals (HG-A);
- Iron Corrosion in Opalinus Clay (IC);
- Long-term Monitoring of Pore-pressures (LP);
- Microbial Activity (MA);
- Mine-by Tests (MB); and
- Full Scale Emplacement Demonstration (FE).

To advance the understanding of the impact of glacial processes on the long-term performance of a DGR, the Greenland Analogue Project (GAP), a four-year field and modelling study of a land-terminating portion of the Greenland ice sheet (2009-2012) located near Kangerlussuaq (Russels Glacier), was established collaboratively by SKB, POSIVA and NWMO. The main objective is to improve the understanding of processes related to groundwater flow and water chemistry adjacent to a continental ice sheet and thereby reduce uncertainties in future safety analyses. The Greenland ice sheet is considered to be an analogue of the conditions that are expected to prevail in Canada and Fennoscandinavia during future glacial cycles. In 2010, significant progress was achieved in the field campaign, which focused on surface water production and routing, radar mapping of ice thickness, ice drilling through to bedrock and water sampling from instrumented boreholes and surface springs. NWMO staff participated in the September field campaign, which focused on surface and groundwater sampling. The NWMO also hosted the Annual GAP workshop and the modelling workshop in November. Specific details on progress in the various GAP sub-projects are summarized in Section 4.3.1.2.

NWMO continued to participate in the international radioactive waste management program of the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA). Members of this group include all the major nuclear energy countries, both waste owners and regulators. NWMO hosted the 20th meeting of the NEA Clay Club (Sept. 21-23, 2010), which was attended by representatives of radioactive waste management programs in France, Switzerland, Belgium and the United Kingdom. The agenda included a presentation of the proposed Kincardine Low and Intermediation Level Waste DGR Project, which was delivered at the Bruce nuclear site rock core storage facility. In addition to the Clay Club, NWMO participated in the following NEA activities:

- Integration Group for the Safety Case (IGSC) Methods for Safety Assessment (MeSA) Project;
- Thermodynamic/Sorption Database development Project;
- Radioactive Waste Management Committee Reversibility & Retrievability Project;
- Radioactive Waste Management Committee; and
- Forum for Stakeholder Confidence.

NWMO continued its participation in BIOPROTA, the international working group on biosphere modelling. At the 2010 annual meeting, NWMO presented work on transfer factors and on glaciation biosphere modelling. Currently, the NWMO is involved in a joint project on U-238 decay chain modelling and will be hosting the 2011 annual meeting.

NWMO is an associate member of the EU FORGE repository gas generation and migration project. This work is co-ordinated with the SKB Äspö LASGIT and Mont Terri HG-A experiments.

2.4 INDEPENDENT PEER REVIEW

The APM Technical Program is reviewed annually by the Independent Technical Review Group (ITRG). In September 2010, the ITRG held their third meeting with the NWMO and reported their findings and recommendations to the NWMO Board of Directors in December 2010. The ITRG 2010 Report and associated NWMO Action Plan can be found on the NWMO website www.nwmo.ca.

3. REPOSITORY ENGINEERING

3.1 INTRODUCTION

The main objectives of the repository engineering program are to: (1) Develop the engineering data, models, methods and tools necessary for the conceptual designs for a deep geological repository (DGR) and associated systems; (2) Provide engineering input to assess the safety of the DGR concept; (3) Support planned site characterization and investigation activities; and (4) Support the development of cost estimates for long-term management of Canada's used nuclear fuel.

In the following sections, the status of the repository engineering program and its achievements in 2010 are outlined for work activities related to research on used fuel integrity (Section 3.2), container corrosion (Section 3.3), repository sealing material development (Section 3.4) and repository design (Section 3.5), including DGR monitoring (Section 3.6), container retrieval (Section 3.7) and transportation of the used fuel to the repository site (Section 3.8). In addition, the NWMO continues to monitor developments on used nuclear fuel reprocessing, partitioning and transmutation (Section 0).

3.2 USED FUEL INTEGRITY

When CANDU fuel bundles are removed from a nuclear reactor, they are stored on an interim basis in wet storage bays for a period of about seven to ten years and then transferred to dry storage facilities at the reactor site. As of June 30, 2010, approximately 2.2 million used Canadian Deuterium Uranium (CANDU) fuel bundles were in storage at Canadian reactor sites (Garamszeghy, 2010). The majority of Canadian used fuel in dry storage is in the Dry Storage Containers (DSCs) developed by Ontario Power Generation.

During the period of dry storage, several mechanisms such as creep rupture, stress corrosion cracking and delayed hydride cracking could potentially affect the integrity of the CANDU fuel bundles. The Used Fuel Integrity Program was established in 2004 to improve the understanding of processes that could affect fuel bundle integrity and to assess whether those processes pose a risk of mechanical failure of the fuel bundles in dry storage. The most important threat to CANDU fuel bundle integrity was found to be Delayed Hydride Cracking (DHC) of the endplate/endcap welds, which could lead to mechanical failure of the bundle.

The Used Fuel Integrity Program completed in 2010 and resulted in publication of three reports (Lampman and Popescu, 2010; Popescu and Lampman, 2010 and Lampman and Gillespie, 2010). A Finite Element Model (FEM) of the CANDU fuel bundle was developed and tested to predict fuel bundle stress fields following its post-in-reactor life (see Figure 3.1; Popescu and Lampman, 2010). This Bundle Stress Model was created with the ANSYS code as a parametric model, which allows the construction of specific models to represent different CANDU fuel bundle types (e.g. 37-element Bruce fuel and 28-element Pickering fuel).

Solutions obtained with the model were verified against test measurements of the deformations experienced by unirradiated commercial bundles subjected to various load conditions. Loads applied to the fuel bundles tested covered both linear and plastic stress regimes. The Bundle Stress Model demonstrated an excellent ability to predict the experimental test data (Popescu and Lampman, 2010).

Concurrent with the development of the Bundle Stress Model, laboratory experiments were completed with unirradiated endplate/endcap welds from artificially hydrided CANDU fuel bundles to hydrogen levels of 40-60 ppm (Shek, 2010). The intent was to obtain a database of threshold stress intensity factors (K_{IH}), which would lead to DHC failure of the welds. The values of this database were then used to compare against the stress intensity factors expected to be operative at the welds as estimated from the field stresses calculated with the Bundle Stress Model.

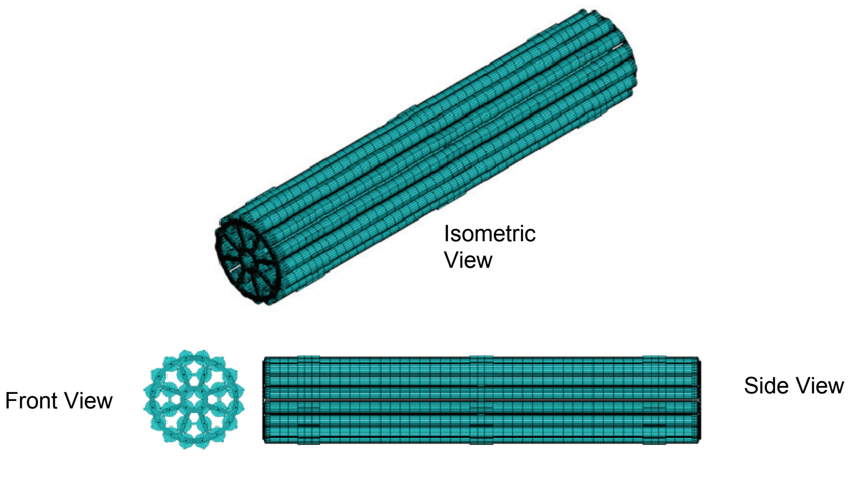


Figure 3.1: Finite element model of the 28-element Pickering CANDU fuel bundle

The results of this evaluation indicated that maximum stress intensity factors on the order of 3.0 MPa $m^{1/2}$ are expected for 28-element Pickering type CANDU fuel bundles when radiation effects are taken into account (Lampman and Gillespie, 2010). The database from the experimental program indicates that the threshold stress intensity factor for DHC crack initiation is in the range of 7.6 to 13.6 MPa $m^{1/2}$. Therefore, DHC is not projected to occur and the bundles are expected to retain their integrity during dry storage. Similar results are expected for the 37-element Bruce type CANDU fuel bundles.

3.3 CONTAINER DEVELOPMENT

3.3.1 Copper Corrosion

The schematic of a copper used fuel container designed for use in a DGR is shown in Figure 3.2. Copper corrosion work in 2010 continued to clarify the mechanism for stress corrosion cracking (SCC) of oxygen free phosphorus doped (OFP) copper. Research activities carried out in 2010 were aimed at improving the understanding of the effects of nitrite and ammonia on the SCC behaviour of copper. Various surface analyses were used to study the fracture and surface oxide of copper, including: (i) X-ray Photoelectron spectroscopy (XPS); (ii) Raman Spectroscopy; (iii) Fourier Transform Infra-Red Spectroscopy (FTIR); (iv) Electron Dispersive X-Ray Analysis (EDX); and (v) Scanning Electron Microscopic (SEM) examinations.

Corrosion experiments concluded SCC was evident in a range of deaerated nitrite concentrations, typically between 0.1 mol/L and 1.0 mol/L, at an applied current density of 1 $\mu A/cm^2$. Lower concentrations of nitrite (≤0.01 mol/L) appeared to be insufficient to induce

SCC. With nitrite concentration ≥ 0.1 mol/L, a 1:1 chloride to nitrite concentration ratio suppressed SCC, as did an overabundance of chloride. High chloride concentrations appeared to increase uniform corrosion, which coincided with a blunting of the crack tip.

In the case of ammonia, the results indicate that SCC was evident at concentrations between 0.3 mol/L and 1.0 mol/L and that an estimated threshold ammonia concentration of 0.3 mol/L is required for SCC. In a 1.0 mol/L ammonia solution, a 1:1 chloride to ammonia concentration ratio suppressed SCC, as did an overabundance of chloride. However, the addition of chloride in concentrations ≤0.5 mol/L had little effect on the SCC behaviour. Detailed results of the experiments will be published in a 2011 NWMO technical report.

Although all the various techniques used to analyse the fracture surface conditions of the copper specimens (specimens pre-exposed to SCC agents) provided useful surface and/or compositional information of the surface oxide, the study concluded that XPS combined with Raman spectroscopy provided the most valuable mechanistic information regarding SCC of copper. Comprehensive results of this surface assessment will be documented in a NWMO technical report to be published in mid 2011.

To support safety assessment, as well as to continue to improve the lifetime prediction of the copper used fuel canister (UFC) in a repository, in 2011 NWMO plans to emphasize corrosion modelling of copper behaviour under postulated repository environmental conditions. The existing suite of copper corrosion models, which were written in customized computer codes, will be converted to COMSOL. The use of COMSOL will facilitate three dimensional modelling representative of different geometries using coupled equations. During the conversion process, it is expected that the latest repository properties and relevant reaction schemes will be implemented to refine the model quality.

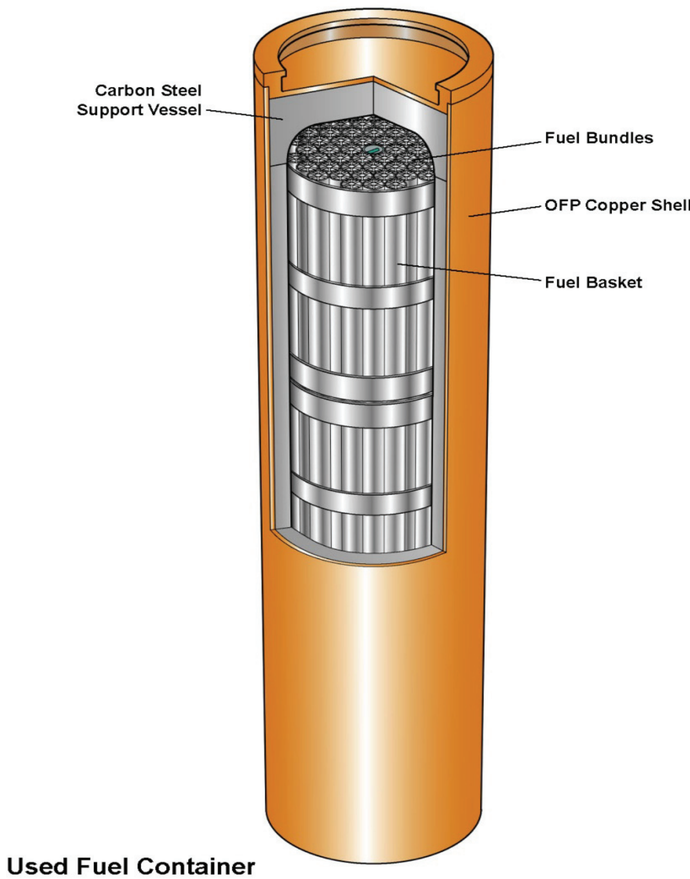


Figure 3.2: Copper used fuel container

3.3.2 Steel Corrosion

In 2010, NWMO continued to advance the design option of a carbon steel used fuel container for use in a DGR (Figure 3.3) and made significant technical advances. With regard to corrosion research, the University of Toronto completed a 3-year study to improve the understanding of carbon steel corrosion under anaerobic conditions. Corrosion tests were conducted on carbon steel wires under different anoxic environmental conditions. The results indicate that corrosion is persistent with massive salt deposition and is likely to stabilize in the range 0.01-0.1 μm/y. However, corrosion becomes extremely slow without salt deposition, even at relative humidities of 75-100% and temperatures of 50-70°C (Newman et al., 2010). Without salt contamination, the corrosion rates are very low, and could only be detected using a solid-state electrochemical hydrogen sensor. The hydrogen sensor can detect pressure increases, as small as, approximately 0.1 Pa, corresponding (depending on the exact procedure) to a corrosion rate as low as approximately 0.0001 μm/y. The estimated corrosion rates for the degreased and pickled wires were found to be < 0.01 μm/y.

In parallel with the corrosion experiments, corrosion product surface analyses were performed. The oxides formed on the carbon steel surface were examined using: (1) Scanning Electron Microscopy coupled with Energy Dispersive X-ray analysis (SEM/EDX) to determine the structure of the corrosion product films; (2) X-ray Photoelectron Spectroscopy (XPS) to identify the chemical composition of the films; and (3) Raman and Fourier Transform Infrared spectroscopy (FTIR) to study bonding. Oxides formed on steel surfaces were found to consist mostly of Fe_3O_4, with some Fe III species from traces of air exposure. Carbonate was detected on the NaCl contaminated surfaces, which had been subject to a degree of prior aerobic corrosion. A high humidity (100%) environment produced more loose surface oxide than a lower humidity environment (75%). Further study findings are detailed in a NWMO Technical Report by Newman et al. (2010).

Figure 3.3: Steel used fuel container

In addition to uniform corrosion study of steel, an extensive literature survey reviewing the probability of stress corrosion cracking (SCC) that might lead to through-wall penetration of a steel UFC in the repository concluded in 2010. Based on environmental and mechanistic evidence, the implications for SCC of carbon steel containers in the repository were considered in the context of the inherent susceptibility of the material, the corrosiveness of the environment and the aggressiveness of the mechanical loading conditions. The survey concluded that the probability of through-wall penetration of the container due to SCC is low. Possible mitigation strategies are also detailed in a NWMO Technical Report by King (2010).

Similar to the copper corrosion model, the existing steel corrosion model, currently written in customized codes, will be converted to a COMSOL platform in order to prepare for future model refinement and expansion.

3.3.3 Container Design

NMWO is studying a number of options for the design of used fuel containers in a DGR. Two possible designs for a copper-shell container and a steel container are illustrated in Figure 3.2 and Figure 3.3, respectively. These particular container designs have different geometries and fuel capacities ranging from 288 to 360 used CANDU fuel bundles. The copper-shell container design has a 25-mm-thick copper shell that provides a corrosion barrier and an inner steel vessel that provides structural support. The steel container consists essentially of a monolithic steel vessel. The containers are closed by a seal weld that secures either the copper lid or the steel lid to the monolithic body. Both containers accommodate a number of steel baskets that maintain the used fuel bundles in a specific geometric array and are designed to withstand an isotropic pre-glaciation pressure load of 15 MPa and a post-glaciation pressure load of 45 MPa. Thermal analysis has indicated that thermally acceptable underground repository designs in crystalline rock and sedimentary rock can be achieved using these container designs.

The dimensions of the type IV-25 used fuel container (both for copper and steel), with a capacity of 360 used fuel bundles, are given in Table 3.1

Over the next few years, NWMO will carry out further development of used fuel container designs, including the assessment of fabrication methods, inspection and seal welding processes. The findings from these studies will be used to identify a reference container design.

Table 3.1: Description of Copper Shell and Steel Used Fuel Container Designs

	Copper-shell IV-25 Container	**Steel IV-25 Container**
Total used fuel bundles per container	360	
Number of bundle layers	6	
Number of bundles per layer	60	
Total weight of 360 used-fuel bundles	8640 kg	
Outer Copper Shell		
- Height (including lid and bottom)	3842 mm	Not applicable
- Outer diameter	1247 mm	
- Inner diameter	1197 mm	
- Wall thickness	25 mm	
- Height of copper lid	110 mm	
- Minimum thickness of copper lid	25 mm	
- Weight (including lid and bottom)	4170 mm	
Steel Vessel Dimensions		
- Height (including lid and bottom)	3700 mm	3880 mm
- Outside diameter	1195 mm	1195 mm
- Inside diameter	990 mm	990 mm
- Wall thickness	102.5 mm	102.5 mm
- Weight (including lid and bottom)	12,650 kg	13,145 kg
Used-fuel baskets		
- Number of baskets per container	3	
- Number of bundles per basket	120 (two layers of 60 bundles)	
- Total weight of 3 loaded baskets	1240 kg	
Total weight of a Container with Fuel	26,700 kg	23,025 kg

3.4 SEALING MATERIAL DEVELOPMENT

In 2010, NWMO continued its program to assess the properties of bentonite-based sealing materials through several laboratory and modelling studies. The work program included: consolidation and triaxial testing; bentonite component swelling studies; and an assessment of as-placed bentonite pellets.

Prior to 2010, triaxial and consolidation tests were carried out on the following sealing materials that are being considered for use in a DGR:

- Highly compacted bentonite (HCB) – 100% bentonite clay;
- Bentonite-Sand Buffer (BSB) – 50% bentonite and 50% sand by mass;
- Dense Backfill (DBF) – 70% crushed granite, 25% lake clay and 5% bentonite by mass;
- Light Backfill (LBF) – 50% bentonite and 50% crushed granite by mass; and
- Gapfill (GF) – 100% bentonite clay fabricated in the form of dense pellets.

In 2010, a triaxial and consolidation program was initiated to assess the properties of a 70/30 bentonite sand mix, which is the reference bentonite material for the APM DGR shaft seal in both crystalline and sedimentary rock.

In addition to the above noted testing programs, the NWMO has also contributed to the **Ba**ckfill and **Clo**sure (BACLO) project. This work was conducted as part of a multi-party (SKB, POSIVA, NWMO) study on piping, erosion and backfill stability done in several laboratories and at the Äspö laboratory in Sweden. As a component of this program, a total of seventeen testing cells were commissioned in order to examine the process of water uptake and volumetric equilibrium of backfill clay blocks in contact with clay pellets. These tests were dismantled at intervals of 3, 9 and 27 months with associated measurement of gravimetric water content and volume changes of the installed components (expansion or compression). Twelve testing cells for 3- and 9-month periods have been dismantled since July 2008 and five testing cells scheduled for 27-months of operation were dismantled in 2010. This information will be useful to provide a better understanding of the role of groundwater salinity on the clay-based sealing materials.

In 2010, NWMO began to assess bentonite pellet preparation for both the in-floor borehole and horizontal tunnel placement methods in order to improve the placement, as well as, the mechanical and thermal properties of the material.

3.5 REPOSITORY DESIGN

The NWMO conducted an evaluation of container placement methods for a DGR based on technical feasibility, safety, siting, monitoring and retrieval for application in crystalline rock, hard sedimentary rock and soft sedimentary rock (Maak et al., 2010). The three generic used fuel container placement methods, in-floor borehole, horizontal borehole and horizontal tunnel, are derived from a review of repository concepts being developed by national radioactive waste management organizations in other countries.

In 2010, the NWMO continued a study to update the generic conceptual designs, implementation schedules and cost estimates for an APM approach that includes a DGR for used fuel in crystalline or sedimentary rock. The update includes associated facilities at the repository site and the systems for transporting used fuel from current storage locations to the repository site.

The base case assumes a used fuel inventory of 3,600,000 used fuel bundles and the alternative case assumes a used fuel inventory of 7,200,000 used fuel bundles. The preliminary requirement is that the packaging and placement rate will be about 120,000 used fuel bundles/year. To evaluate the sensitivity of the cost estimate for the APM transportation system and DGR to variances in host rock type and placement method, conceptual designs and cost estimates are being finalized for DGR options at a depth of 500m:

a) In crystalline rock using the in-floor borehole placement method; and
b) In sedimentary rock using the horizontal tunnel placement method.

An example of the conceptual repository level layout for the crystalline rock geosphere is shown in Figure 3.4. Features of the repository level include:

(1) A footprint of approximately 2.6 by 1.6 km;
(2) 124 placement rooms spaced at 40 m, and designed to hold a total of 10,000 UFCs;
(3) An underground demonstration facility, near the main and service shafts, that will be used to demonstrate prototype equipment and conduct geosphere confirmation studies; and
(4) Interconnected drifts and cross-cuts to support: the transport of excavated rock; equipment; UFCs and backfill materials; and the underground ventilation system.

In addition to the APM design and cost update, NWMO is supporting the dismantling of the Prototype Repository at the Äspö laboratory in Sweden and initiating a scale trial of the horizontal tunnel placement method. NWMO is also continuing to model aspects of the underground repository, including near-field modelling of the UFC and a shaft seal.

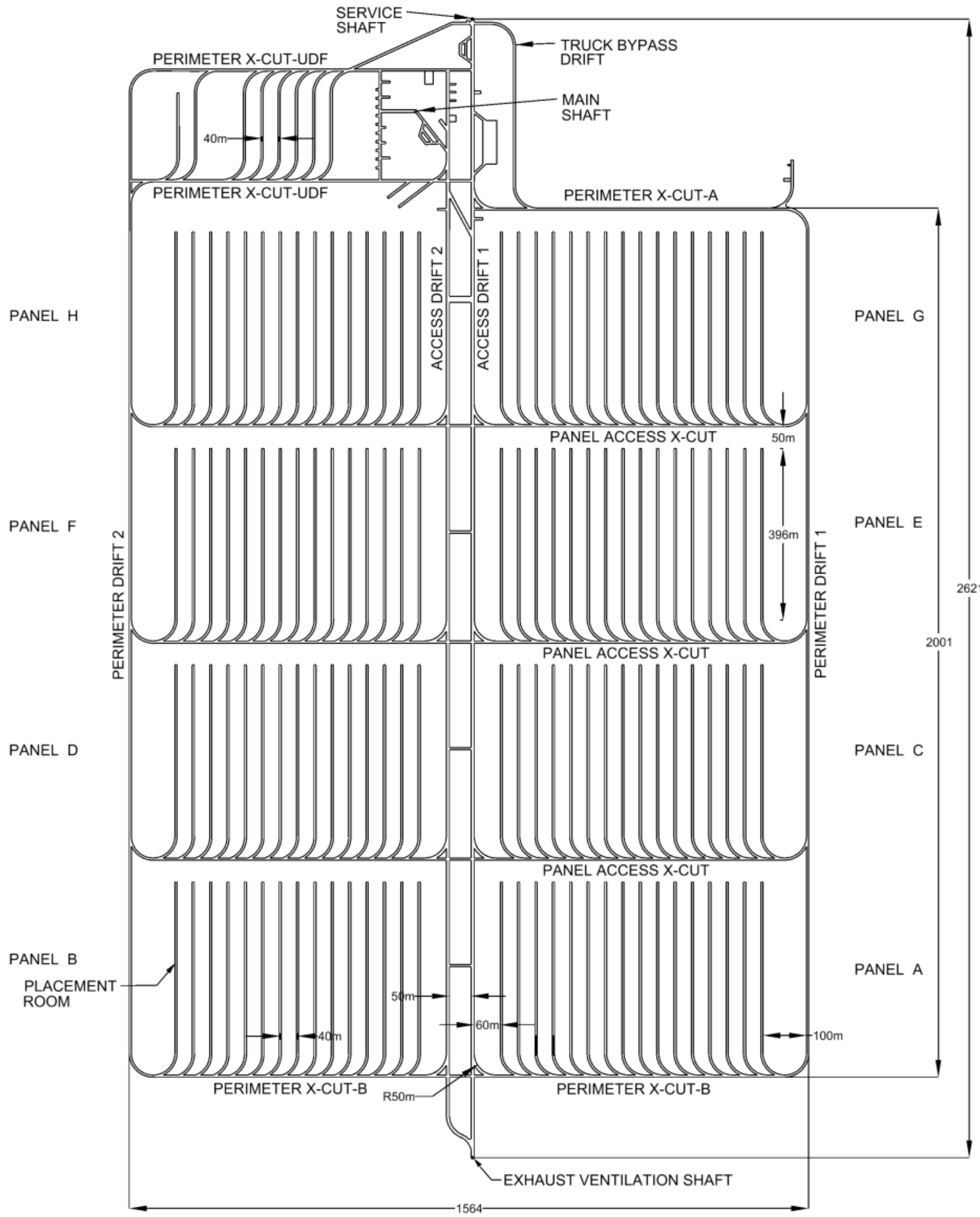


Figure 3.4: Conceptual underground demonstration facility (UDF, top left) and repository layout for crystalline rock. All distances are given in meters

3.5.1 Numerical Modelling of a Deep Geological Repository

A series of three-dimensional thermal transient and thermal-mechanical (T-M) stress analyses was performed on a DGR for used CANDU fuel using the horizontal tunnel placement geometry at a depth of 500 m in limestone (Guo, 2010). Based on the near-field modelling, and as shown in Figure 3.5, the peak temperature of the container surface is 117.0˚C, 10 years after UFC placement and the peak temperature of the tunnel surface is 69.0˚C, 50 years after used fuel placement.

Coupled T-M far-field analyses were used to determine the peak temperatures at various regions in the repository (Guo, 2010). The peak temperature in the rock is 42.7˚C at the centre of the repository after 1,200 years. The peak temperatures at the centre of the repository edge (727.5 m from repository centreline) and repository corner (1,265 m from repository centreline) are 27.6˚C and 20.3˚C, respectively, 4,000 years after placement. These analyses predicted that the maximum thermally induced uplift at the ground surface above the centre of the repository would be about 0.13 m. This degree of deformation was determined to be insufficient to generate additional fractures in the rock near the ground surface.

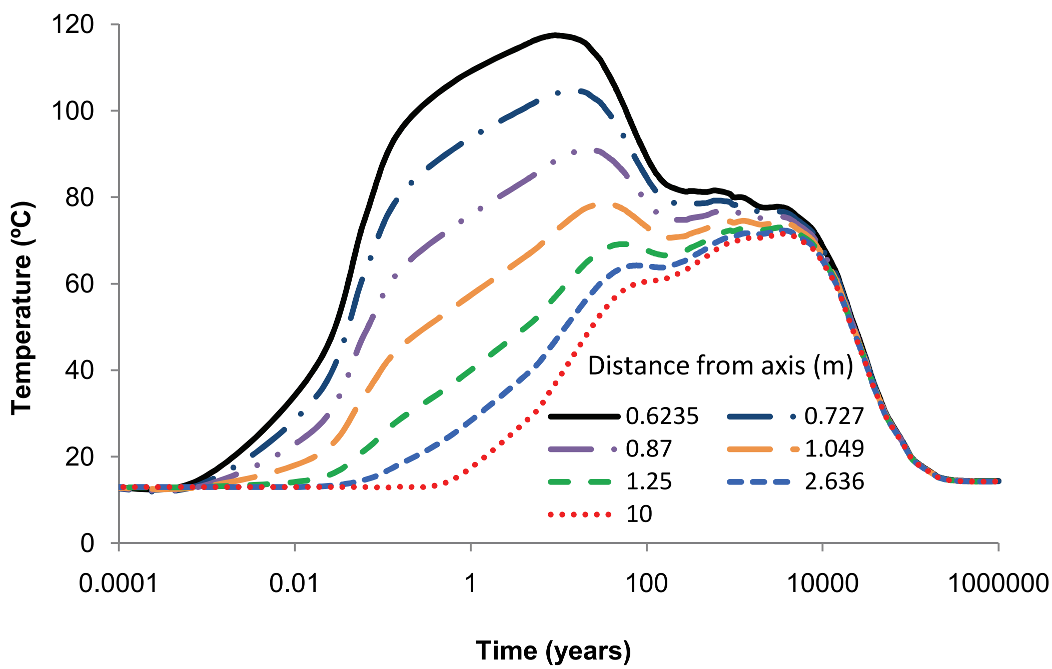


Figure 3.5: Temperatures at different locations along a horizontal line extending outward from used fuel container

3.6 MONITORING

Monitoring of the DGR is an integral component of APM through the various phases of implementation. Consultations with Canadian stakeholders have indicated strong support for the concept of continuous monitoring of the used nuclear fuel repository and the repository site over an extended period of time. Within the context of APM, monitoring can be sub-divided into two major categories:

a) Monitoring of processes and parameters in the living environment and geosphere during site characterization and operation; and

b) Monitoring of DGR systems once used nuclear fuel has been placed in the repository to confirm that:
   i) The repository systems are performing their function as expected; and
   ii) The conclusions of the safety analyses regarding the capability of the repository system to safely manage used nuclear fuel over the long-term remain valid.

Monitoring of environmental and geosphere parameters during the site characterization stage will provide baseline information for site evaluation, environmental assessment, and a licence application for a DGR at the preferred repository site. Site characterization studies will inform repository preliminary design and safety assessment studies for the DGR and will serve to identify required monitoring functions specific to the site and site-specific design. The monitoring system will evolve along with the DGR systems design and will fulfil different functions along the different APM implementation phases.

3.6.1 Enhanced Sealing Project

Two shaft seals were installed in 2009 at the URL, one in the access shaft and one in the ventilation shaft, as part of the facility decommissioning process, for the purpose of isolating the deep saline groundwater systems from near-surface low-salinity groundwaters. The seals were placed at the intersection of each shaft with a shallow fracture zone (FZ2), as shown in Figure 3.6. In 2008, AECL proposed a three year project to instrument the access shaft seal and monitor the early evolution of the seal components. NWMO, ANDRA (France), POSIVA (Finland) and SKB (Sweden) agreed to participate in the project, which was designated as the Enhanced Sealing Project (ESP). The scope of the ESP included the design and installation of a suite of instruments in the access shaft seal, and monitoring of seal evolution over a three year period.

The installed access shaft seal consists of a lower, reinforced low-heat high performance concrete component designed to act as structural support for the entire seal, a 40/60 clay-sand mixture component (compacted in situ) of approximately 5 m diameter and 6 m in length, and an upper concrete component intended to restrict expansion of the clay component. Upon saturation, the clay component is expected to develop a swelling pressure of approximately 2 MPa.

A full description of the seal and the instrumentation array is given by Dixon et al. (2009). The ESP instrumentation includes a suite of 66 instruments monitoring 95 seal parameters, including strain, temperature and hydraulic pressure in the concrete components, water content, pore pressure, total pressure and temperature in the clay component. Total displacement is also monitored at the top of the upper concrete component.

Two reports prepared by AECL document the design, construction and instrumentation of the seal and provide initial monitoring results (Martino and Katz, 2010; Dixon et al., 2009). Early monitoring results, reported in 2010, indicate that the seal components are behaving as predicted. The major remaining task in the project is the continued collection and interpretation of the data. The ESP current scope of work will be completed at the end of 2013.

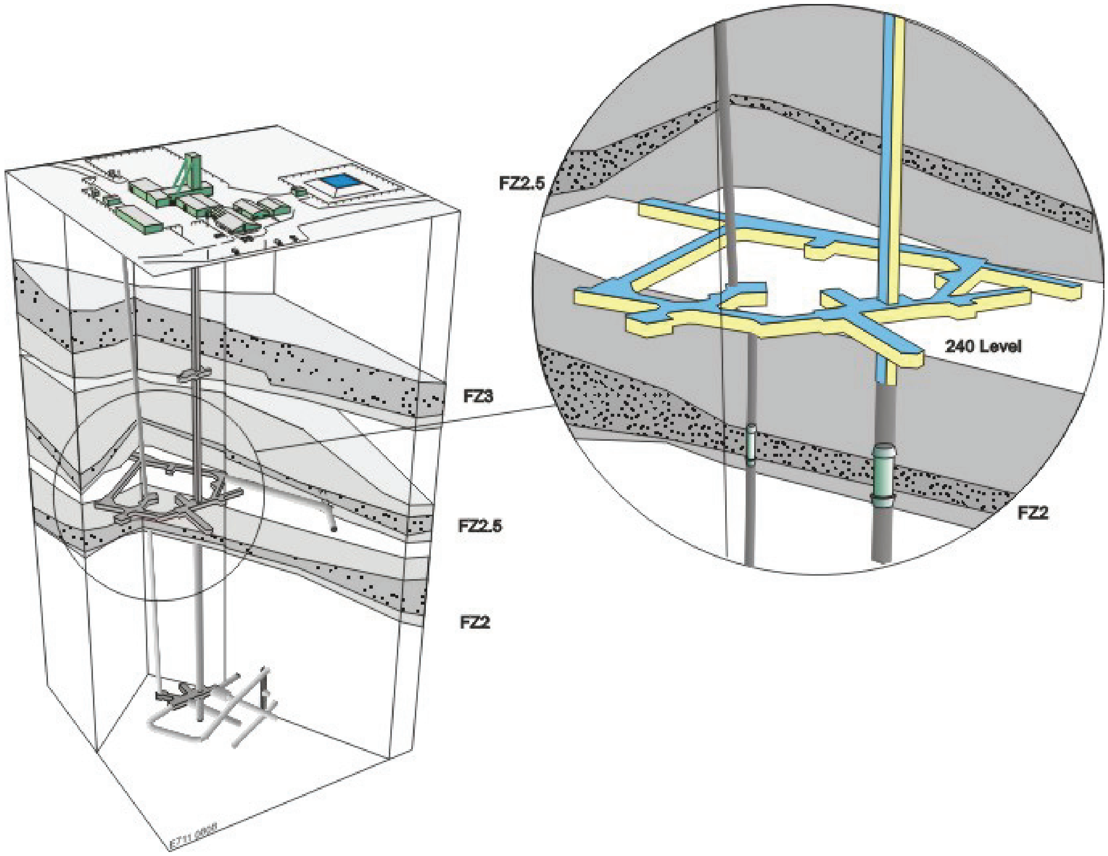


Figure 3.6: Configuration of the AECL Underground Research Laboratory. Inset shows the location of the instrumented Enhanced Sealing Project seal installed where the access shaft (on the right side) intersects Fracture Zone 2 (FZ2), at a depth of approximately 270m

3.7 RETRIEVAL

Retrievability of used nuclear fuel throughout all phases of implementation is a fundamental feature of APM (NWMO, 2005). Since 2008, the NWMO has been participating in the Nuclear Energy Agency (NEA) Working Group on Reversibility and Retrievability (R&R). The key objective of the NEA working group is to provide various NEA member countries a forum to share their experience and resources in the area of nuclear waste retrieval.

In 2010, the NWMO participated actively in the R&R working group and provided significant contributions and input in developing the R&R project report (to be published by NEA). This report documents the views expressed by representatives of national waste management organizations and summarizes the findings obtained from dialogues within the waste management community.

In addition to participating in international discussions and activities, NWMO continues the development of conceptual designs for used fuel container retrieval systems for both crystalline and sedimentary rock repository designs. Preliminary design concepts of the retrieval systems will be published in 2011.

3.8 USED FUEL TRANSPORTATION

The short-term objectives of the used nuclear fuel transportation program are to: develop and evaluate options for the transport of used fuel from interim storage sites to a used fuel DGR site; and provide engineering design and logistics to support the evaluation of potential candidate repository sites. As feasibility studies are being completed on potential host communities interested in the APM program, the transportation program objective is being addressed through the development of conceptual designs and illustrative case studies. In the longer term, the transportation program will implement the transport of used nuclear fuel to the DGR site.

To provide an up-to-date understanding of used nuclear fuel transport, Stahmer (2009) reviewed and documented the state of programs within Canada and in several countries that are actively working in the transport of used nuclear fuel. Advances in the transportation system conceptual design, logistics and risk assessment are detailed in Sections 3.8.1 to 3.8.3.

3.8.1 Transportation System Conceptual Design

Used fuel transportation is included in the updated APM engineering design and cost studies and is an integral part of Canada's plan for the long-term management of used nuclear fuel. In 2010, a conceptual study including the loading of transportation packages at each of the current used fuel storage sites followed by all-road transport to a hypothetical APM facility was prepared and will be published in 2011. The costs associated with these activities were calculated separately. Future studies will assess the transport of used fuel using all surface transport modes, including road, rail and water.

3.8.2 Transportation System Logistics

In May 2010, the APM siting process was launched. Since then, a number of communities have expressed interest in hosting the APM DGR. High-level infrastructure and logistics studies identifying the existing transportation infrastructure that links the existing fuel storage sites and the potential host communities are being considered for all surface modes of transport.

3.8.3 Transportation Risk Assessment

As part of APM implementation, an update to the generic risk assessment for the transport of used nuclear fuel from the interim used fuel storage sites to a hypothetical repository site located in Canada was initiated. This risk assessment is being updated to incorporate current system design assumptions and will support community engagement addressing public concerns about the safety of the transportation system and its associated risks.

The previous risk assessment of the used fuel transportation system (OHN, 1994) examined three hypothetical repository sites, different modes of transport and a set of accident conditions specified in the transportation regulations.

The risk assessment update in 2011 will examine the risks associated with normal transport and accident conditions as defined within the transportation regulations. Additional transportation studies will be considered following the updated to the generic risk assessment.

3.9 REPROCESSING AND ALTERNATIVE WASTE MANAGEMENT TECHNOLOGY

NWMO continues to monitor developments on used nuclear fuel reprocessing, partitioning and transmutation (RP&T) as alternative technologies that could be used for long-term management of nuclear fuel waste. In 2010, the third annual watching brief on used fuel and alternative waste management technologies was published (Jackson and Dormuth, 2010).

The watching brief reviewed international developments by examining recent publications and presentations given at the Nuclear Energy Agency information exchange meeting on RP&T (Nov. 2010, San Francisco). The U.S., several European countries and Japan are in the process of reviewing their RP&T programs. In the U.S., a Blue Ribbon Commission was appointed for the review and will set the future direction of RP&T in their country. Both France and Japan have established 2012 as a major decision point, at which time, the future direction of these two leading RP&T programs will be decided.

On the basis of the three watching briefs (Jackson and Dormuth 2008; 2009; 2010), the current status of RP&T was evaluated. It was determined that despite the progress in R&D on advanced closed fuel cycles, they remain many decades away from commercial deployment and will require a very large investment in nuclear infrastructure. In addition, closed fuel cycles will not eliminate the need for a DGR. As such, Jackson and Dormuth (2010) concluded that at this time, there is no compelling reason to alter the reference APM strategy of a DGR for used CANDU fuel.

In addition, Jackson and Dormuth (2010) continued to monitor progress in deep borehole disposal concept and discussed implications of that progress in the Canadian context. Assessment of the deep borehole disposal for light water reactor (LWR) fuel by Sandia Laboratories (Swift, 2010), suggests long-term performance is likely to be excellent. Due to the lower reactivity and burnup of CANDU fuel relative to LWR fuel, the performance case for LWR fuel is conservative in the Canadian context. However, retrievability (a key component of APM) will be very difficult, if not impossible, to implement with the deep borehole disposal method. In addition, cost effectiveness for used CANDU fuel has not yet been demonstrated.

4. GEOSCIENCE

4.1 INTRODUCTION

The main objectives of the NWMO's geoscience program are to: (1) Evaluate the adequacy of potential candidate sites for a deep geological repository (DGR) by conducting site characterizations; and (2) Advance the understanding of geosphere stability and its resilience to long-term perturbations.

The geoscience program is developing plans and methods to assess the suitability of potential candidate sites in willing host communities and refining the understanding of geosphere processes related to the long-term stability and performance of a DGR. This is achieved through a multidisciplinary approach involving the coordinated effort of research groups drawn from Canadian universities, consultants, federal organizations and international research institutions. In particular, the geoscience program is a partner and participant in the Äspö Modelling Taskforce, Greenland Analogue Project and Mont Terri Underground Rock Laboratory Project.

The following sections outline the activities of the geoscience work program in 2010. The activities are organized into sections aligning with the objectives of the program to develop site characterization plans and methods (Section 4.2) and further understanding of long-term geosphere stability (Section 4.3).

4.2 METHODS FOR SITE INVESTIGATION

In 2010, NWMO issued "*Moving Forward Together: Process for Selecting a Site for Canada's Deep Geological Repository for Used Nuclear Fuel*" (NWMO, 2010). The document includes a list of site evaluation criteria and describes the site evaluation process. The suitability of potential candidate sites to safely host a DGR will be evaluated over many years in a stepwise approach, consisting of: a) initial screenings to evaluate the suitability of candidate sites against a list of initial screening criteria, using readily available information; b) feasibility studies to further determine if candidate sites may be suitable for developing a safe used fuel repository; and finally c) detailed field investigations to confirm suitability of one or more sites based on detailed site evaluation criteria. Each step is designed to evaluate the site in greater detail than the previous step. More information can be found at http://www.nwmo.ca/sitingprocess.

In 2010, NWMO also continued to develop readiness for detailed site investigations of willing host communities during the site selection process. NWMO continued to develop and refine methods for characterization of geochemistry (Section 4.2.1), radionuclide transport (Section 4.2.2), microbiology (Section 4.2.3), geomechanical properties (Section 4.2.4), seismicity (Section 4.2.5) and hydrogeology (Section 4.2.6).

4.2.1 Matrix Pore Water Extraction and Geochemical Analysis

Chemical and stable isotopic (e.g., $\delta^{18}O$, $\delta^{2}H$) compositions of groundwaters and matrix porewaters provide information on the origin and evolution of the groundwater system and can be used to determine groundwater fate over geologic time frames. In addition, near-field performance, safety assessment and groundwater transport/evolution models require knowledge of groundwater geochemical compositions. The composition of water within the rock matrix of crystalline and sedimentary formations may have compositions similar to those of the groundwaters. However, direct information on porewater composition is required in order to support this hypothesis. Extraction of porewater from low-permeability sedimentary rocks with low water content can be challenging. In 2010, the NWMO continued to explore the applicability of ultracentrifugation, vacuum distillation, crush and leach, and an extraction method using capillary action for extraction of porewater from low permeability sedimentary rocks. In addition, NWMO further developed the diffusive exchange technique for measuring isotopic compositions of saline waters.

Assessment of the feasibility of the ultracentrifugation method for site characterization applications in sedimentary and crystalline rock began in 2005 in collaboration with Gascoyne GeoProjects and the U.S. Geological Survey (Gascoyne and Hobbs, 2009) and continued in 2010. Experimental results to date indicate that the ultracentrifugation technique may not be able to extract matrix porewater from very low porosity samples. However, results show good reproducibility between replicate analyses of moderate porosity sedimentary rocks. In addition, good reproducibility was observed during successive extractions from the same limestone core sample. In 2011, porewater extraction method development will focus on vacuum distillation and crush and leach methods in collaboration with the University of Ottawa. In addition, an extraction method using capillary action to extract small quantities of porewater onto absorbent, low chemical-background papers sandwiched between rock core segments will be explored with the University of New Brunswick.

A two-year work program with the University of Bern to evaluate and benchmark a newly adapted diffusive exchange technique for measuring stable isotope compositions of saline matrix porewaters continued in 2010. Testing of this technique focuses on highly saline porewaters, such as those found in Canadian crystalline and sedimentary rocks. The objectives of the on-going work program are to: (1) Conduct testing to evaluate and benchmark the adapted diffusive exchange technique and determine its applicability to rocks with a range of different mineralogies, porosities and porewater salinities; and (2) Improve our ability to constrain anion-accessible porosities in rocks containing highly saline porewaters. This information is important for the interpretation and modelling of porewater evolution – for example, from natural tracer profiles collected across sedimentary sequences.

4.2.2 Assessment of Radionuclide Transport Processes

A key issue for DGR performance assessment of the long-term containment of used nuclear fuel is the understanding of dominant solute transport processes of radionuclides. Within very low hydraulic conductivity crystalline or sedimentary formations, transport processes are dominated by diffusion. Sorption of solutes onto mineral surfaces is another important mechanism for retarding radionuclide transport. Hence, measurement of diffusion and sorption properties is important for demonstrating and illustrating the containment and isolation capabilities of low permeability rocks, and for increasing confidence in the DGR safety case. NWMO work on diffusion and sorption in 2010 is detailed in Sections 4.2.2.1 and 4.2.2.2, respectively. In addition, NWMO evaluated the potential for colloidal-bound transport of radionuclides as described in Section 4.2.2.3.

4.2.2.1 Diffusion

Diffusion is considered to be the dominant mechanism for solute transport in low permeability geological formations. The diffusion coefficient is an important parameter for numerical simulation of radionuclide transport. Previous NWMO work programs successfully developed an X-ray radiography technique to quantify diffusion coefficients using conservative iodide and tritium tracers in collaboration with the University of New Brunswick (Cavé et al., 2009a, 2009b).

The technique was expanded to quantify diffusion-reaction processes using a non-conservative tracer, cesium (Cavé et al., 2010). This technique was then applied to investigate the reactive transport of non-conservative solutes in a sorption and diffusion dominated sedimentary rock system (Cavé et al., 2010). Reactive transport modelling, coupling diffusion and ion-exchange, matched experimental data and was successfully used to quantify the cation exchange capacity of intact rock samples.

Cavé et al. (2010) also established a protocol for characterizing the variability in diffusive properties across a sedimentary rock formation using the relationships between diffusion and lithology to assess the scale-dependency of diffusive transport. Investigation of the potential for up-scaling of laboratory diffusion measurements is currently being investigated through the comparison of measured laboratory-scale diffusion coefficients to in situ field-scale diffusion coefficients made under the Mont Terri radionuclide diffusion and retention (DR) long-term experiment. Additional details on the Mont Terri project and published reports can be found at the Mont Terri Project website (http://www.mont-terri.ch).

4.2.2.2 Sorption

Sorption is an important mechanism for retarding the transport of radionuclides to the biosphere. On the basis of the recommendations of a state-of-the-science review conducted in 2008 (Vilks, 2009), a two-year experimental program on sorption was initiated with Atomic Energy of Canada Limited (AECL) in 2009 to develop an understanding of sorption processes in Na-Ca-Cl brine solutions with Canadian sedimentary rocks. Experimental protocols for performing batch sorption experiments in brine solutions were developed and batch sorption experiments were performed with Canadian sedimentary rocks (shale and limestone), bentonite, and various saline solutions representative of groundwater compositions in Canadian sedimentary rocks with TDS values as high as 300 g/L.

Sorption measurements of Sr (as an analog to Ra), Cu, Ni (as an analog for Pd, Pb and other transition elements), Eu (as an analog for trivalent actinides) and U were conducted. Results confirmed that elements such as Sr, that attach to minerals mainly by non-specific Coulombic attraction, do not sorb in brine solutions, whereas transition elements, such as Cu and Ni, trivalent Eu and actinides U, sorb in brine solutions by surface complexation mechanisms. The experimental work program investigated the impact of salinity and pH on sorption processes for these elements. Sorption kinetics studies were performed to determine the sorption equilibrium time and the sorption reversibility.

As part of the 2009-2011 work program, the feasibility of performing mass transport experiments in a rock matrix was tested using the High Pressure Radioisotope Migration (HPRM) apparatus (Vilks and Miller, 2007). The initial experiments indicate that mass transport experiments through unfractured shale is possible but is also very time-consuming due to the very low permeability of the samples of interest as DGR host/cap rocks. Columns of crushed rock, or rock samples with fractures, could be applied for future dynamic mass transport experiments.

In addition, international literature and sorption databases were reviewed to find available sorption data relevant to Canadian sedimentary rocks (shale and limestone) and bentonite in a setting that includes Na-Ca-Cl brine solutions at near neutral pH. A set of recommended sorption values for twenty elements (C, Cu, As, Se, Zr, Nb, Mo, Tc, Pd, Pa, Sn, Pb, Bi, Ra, Th, Pa, U, Np, Pu and Am) was identified for bentonite, shale and limestone. These sorption values provide the starting point for the development of a Canadian sorption database for sedimentary rocks.

4.2.2.3 Colloidal transport

In 2010, the NWMO work program on colloids was directed towards the preparation of a summary report of the work performed to improve the understanding of the role of colloids on the performance of a DGR. The report will be issued in 2011 in collaboration with SKB. The report includes a summary of the results from NWMO's colloid transport and erosion experiments along with SKB's Colloid Dipole and Colloid Transport projects (2005-2010), which were focused on: (1) Quantifying natural colloid concentrations as a function of ionic strength; (2) Improving the understanding of colloid stability, transport, and generation from bentonite; and (3) Colloid facilitated transport of actinides.

The current understanding of colloids indicates that natural colloid concentrations are too low to have a significant impact on radionuclide transport. Radionuclide-enriched colloids generated from the waste form are likely to be trapped by the engineered barrier. Erosion tests examining the effect of water composition, bentonite composition, fracture dip and fracture aperture suggest that in waters containing millimolar amounts of dissolved salts (representative of glacial melt water) bentonite will expand into an open fracture (Vilks and Miller, 2010). It is expected that this will likely form stable deposits that do not readily erode or release significant concentrations of colloids (Vilks and Miller, 2010). Any montmorillonite colloids released are expected to be in amounts that are too small to compromise the integrity of the buffer, but could be radionuclide-enriched. However, in most groundwater, montmorillonite colloids are not stable and are likely to agglomerate and be filtered over short transport distances. Even in dilute waters the likelihood of stable colloids being transported over significant distance is very small due to low natural water velocities, and the ability of the geosphere to trap colloids (even those with the same surface charge as most minerals). In the case of fracture pathways, surface irregularities, heterogeneity in surface charge distribution, and fracture aperture variability, contribute to colloid retention.

4.2.3 Microbiology Processes & Methods

NWMO microbiology research in support of the development of a DGR for used nuclear fuel investigates the role of microorganisms in both the near- and far-field as discussed in Sections 4.2.3.1 and 4.2.3.2, respectively. The near-field program examines activity and survivability in bentonite clay buffer materials, while the far-field program examines the microbiology of the host rock. The objective of these research programs is to identify design provisions that may be necessary to suppress microbial activity in the repository and to prepare for site characterization activities.

4.2.3.1 Near-field microbiology

One of the key performance requirements of the repository sealing system is to suppress microbial activity at or near the used fuel container surface in a DGR. Microbial investigations at the AECL Underground Research Laboratory have found that heterotrophic aerobes, anaerobes and sulphate-reducing bacteria are present in the buffer, backfill and sealing materials studied. These studies also indicated that 100% highly compacted bentonite (HCB) directly in contact with the UFC's is necessary to reduce microbial activity to insignificant levels (Stroes-Gascoyne, 2010).

In 2010, a series of experiments assessing the effects of bentonite dry density, salinity and temperature on microbial activity were conducted with 100% HCB to identify repository conditions and design provisions that might be required to suppress microbial activity and prevent microbiologically influenced corrosion (MIC) of copper containers in a DGR (Stroes-Gascoyne et al., 2010a; Stroes-Gascoyne et al., 2010b). The results suggest that in a low-salinity repository environment, a HCB with a dry density of 1.6 g/cm^3 will suppress microbial aerobic culturability below background levels (2.1 x 10^2 colony-forming units/g) in as-purchased bentonite (Stroes-Gascoyne et al., 2010b). In contrast, in a high-salinity repository environment, with porewater salinities ≥ 50 g/L, salinity will suppress microbial activity over a wider range of bentonite dry densities (~ 0.8 to 1.8 g/cm^3) and a wide range of bentonite swelling pressures (Stroes-Gascoyne et al., 2010a, Stroes-Gascoyne et al., 2011a). Experiments assessing the effect of temperature suggest that cells were not particularly sensitive to a temperature of 60°C and some culturability remained after exposure to 80°C at all dry densities studied (0.8 to 2.0 g/cm^3), whereas at temperatures ≥ 121°C, culturability was reduced. Importantly, the effect of temperature on culturability in low dry density bentonite was reversible once the heat source was removed and re-saturation was allowed to occur, highlighting the importance of maintaining high dry density to keep microbial activity to a minimum (Stroes-Gascoyne and Hamon, 2010).

In 2011, a State of Knowledge review regarding near-field microbiological considerations relevant to a DGR for used nuclear fuel will be conducted by researchers at Ryerson University and University of Saskatchewan. On the basis of the review, a path forward for the NWMO's near-field microbiology program will be established that is consistent with the approaches taken by other international nuclear waste management organizations.

4.2.3.2 Far-field microbiology

In 2010, NWMO initiated a State of Knowledge review regarding far-field microbiological considerations relevant to a DGR for used nuclear fuel with the University of Toronto. The review evaluates the implications of in situ microbiology on geochemical conditions, radionuclide migration and gas production and summarizes the microbiology research conducted by the Canadian program and internationally. The review will be published in 2011 and will outline a preliminary approach for microbial analyses that should be conducted as part of detailed site characterization activities.

The Canadian microbiology research program in support of the development for a DGR for used nuclear fuel has evaluated the in situ microbial communities present in crystalline (Stroes-Gascoyne et al., 2001 and references therein) and sedimentary geological environments (Stroes-Gascoyne and Hamon, 2008). In 2010, NWMO continued its involvement in the Microbial Activity (MA) experiment at the Mont Terri Underground Research Laboratory (Stroes-Gascoyne et al., 2011b; Wersin et al., 2011). As part of the project, borehole water and Opalinus Clay overcore samples from around the Porewater Chemistry (PC) experiment borehole were analyzed using molecular biology and culturing microbiological methods. The results of the microbial analyses highlight the importance of anaerobic microbial activity in "disturbed" Opalinus Clay, as facilitated by the introduction of space, water and organic material (Wersin et al., 2011). The resulting anaerobic microbial activity had an effect on pH and Eh. However, the results also suggest that such effects were temporary and spatially limited because of the large buffering capacity and diffusive properties of the clay. In 2011, NWMO will continue its participation in the MA experiments at Mont Terri. This will facilitate further experience in microbial site characterization methods and understanding of microbial processes in a DGR for used nuclear fuel.

4.2.4 Geomechanical Methods & Assessment

NMWO's geomechanics program is evaluating the natural state of rocks and determining their mechanical response to imposed loads and/or deformations. Assessment of such thermo-hydro-mechanical (THM) processes in rocks is increasingly important due to the demands for the design, construction, operation and performance/safety assessments of underground used nuclear fuel repositories, and other civil and environmental engineering works. In general, the properties of the host rock will be altered by the THM impacts of the used fuel. As such, gathering reliable rock mechanical data is required for the modelling and engineering design of the underground openings at various scales.

Understanding the geomechanical properties of potential DGR host rocks is an important component of NWMO's R&D program. In 2010, an experimental work program at the University of Toronto, initiated in 2009, to characterize THM coupled processes and properties continued (Section 4.2.4.1). In addition, the suitability of FRAC3DVS-OPG, to simulate a regional scale fractured rock mass was further explored (Section 4.2.4.2). To further understand the presence and distribution of fractures in intact rock masses and processes affecting the excavated damage zone (EDZ), NWMO also continued its fracture network (Section 4.2.4.3) and EDZ (Section 4.2.4.4) modelling programs, respectively.

4.2.4.1 Measurement of THM properties

The objective of the University of Toronto THM experimental work program is to test thermal properties of sedimentary rocks that are representative of those found in Canadian deep sedimentary basins. In 2010, two sets of experiments were carried out. The first set involved uniaxial compressive strength (UCS) tests (Figure 4.1) on dry and saturated Lindsay limestone samples. The tests were conducted with samples heated up to 100°C. The results from the UCS tests suggest that the Lindsay limestone is generally fairly strong, with an average uniaxial compressive strength of 121±21 MPa for the dry samples. Average saturated UCS for the rock is 83±18 MPa. Reported errors are 1σ standard deviations. Heating up to 100°C did not have a detectable effect on strength and deformational properties of the rock samples.

(a) (b)

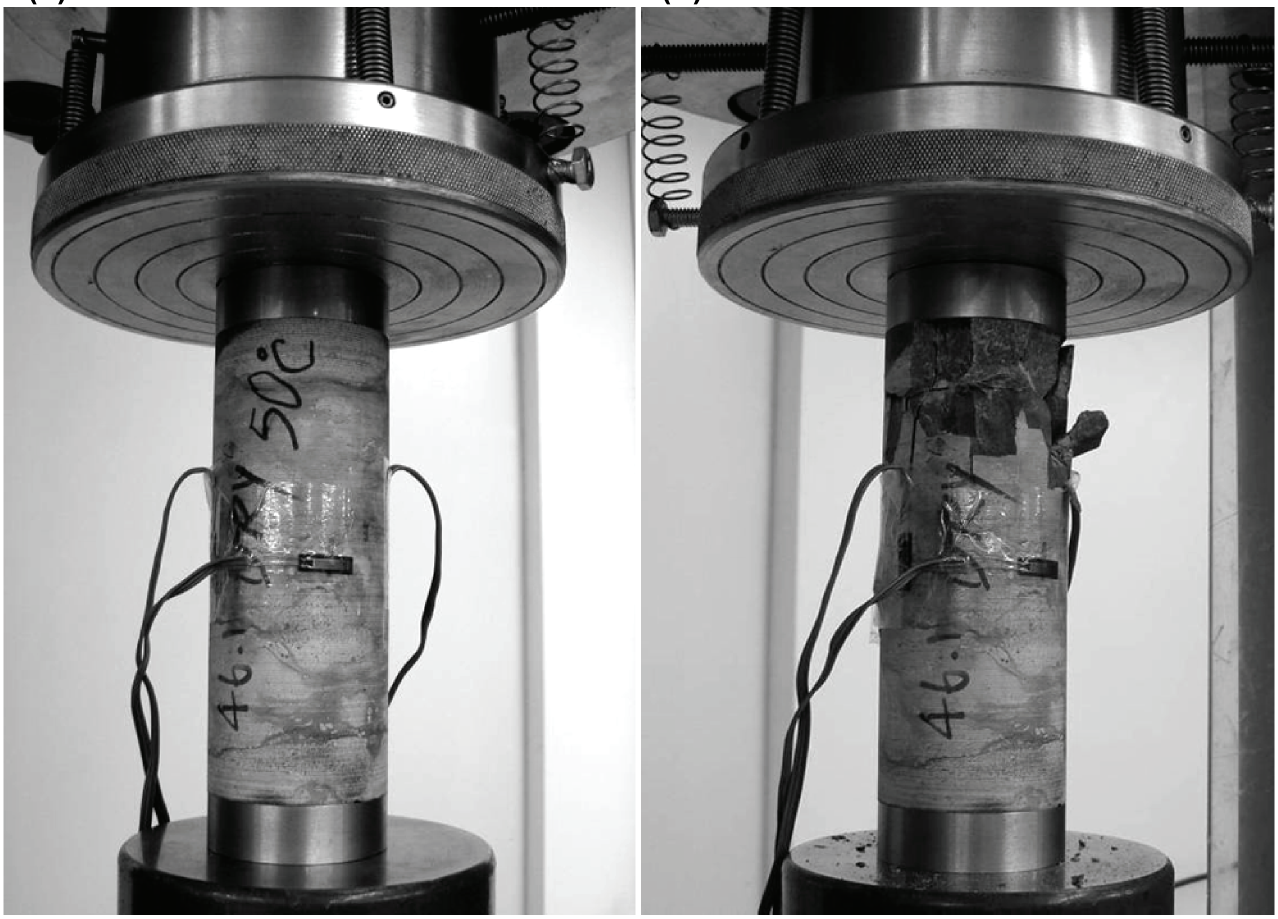


Figure 4.1: Uniaxial compressive strength test on dry Lindsay limestone. (a) Before testing, the sample is attached with strain gauges. (b) After testing, showing the failed sample exhibiting spalling. Rock cores shown are 135 mm in length and 54 mm in diameter

The second set of experiments used a servo-controlled permeameter to measure ultrasonic wave velocity and permeability changes with temperatures up to 125°C at hydrostatic pressures up to 12.5 MPa. The results suggest that the permeability of the samples decreased with the increase in hydrostatic stress and with temperature. Seismic wave velocity evolution and strain measurements further confirmed this decrease. Phase 2 of the experimental work commences in 2011 and includes tests to study the effect of confinement on mechanical and hydraulic properties of the Lindsay limestone.

4.2.4.2 Hydro-mechanical enhancement of FRAC3DVS-OPG

A host rock for a potential DGR will be subjected to variable stress conditions over the course of the repository lifetime. These stresses include the in situ stress of the rock, stress induced by excavation of the repository, thermo-mechanical stress due to heat from the containers and glacial stress. In the event of glacial loading, up to 3 km of ice may be present on top of the rock mass. This glacial stress is transmitted into the geosphere and will impact both the rock mass, as well as fluids in it.

In a fractured crystalline environment glacial loading could cause changes to the hydraulic properties of the rock mass, in both the unfractured rock matrix and along the fracture planes. These hydraulic changes can potentially alter groundwater flow patterns, thereby affecting radionuclide transport. The effect of the glacial loading is not limited to the rock mass directly under the ice. A change in the stress can also be observed ahead of the glacial front, caused by the Poisson effect.

FRAC3DVS-OPG is a numerical modelling code used to simulate groundwater flow and solute transport. Application of FRAC3DVS for hydrogeological simulations is discussed in Section 4.3.2.1. Hydro-mechanical coupling in FRAC3DVS-OPG is limited to purely vertical strain with lateral constraints (Guvanasen, 2007). In order to better address the issue of thermo-hydro-mechanical coupling, a new coupling module has been developed and incorporated into the FRAC3DVS-OPG code. The coupling approach is based on the extended Biot's formulation for non-isothermal consolidation in poroelastic materials. The module has been verified against known analytical solutions for isothermal and non-isothermal consolidation. Fractures are incorporated into the module as stress-dependent composite material properties. In 2010, the module was applied to simulate a hypothetical arrival of a glacial front. Preliminary results indicate that a hydro-mechanical response to an approaching glacier may be felt at distances far away from the leading edge of the glacier.

4.2.4.3 Fracture network modelling

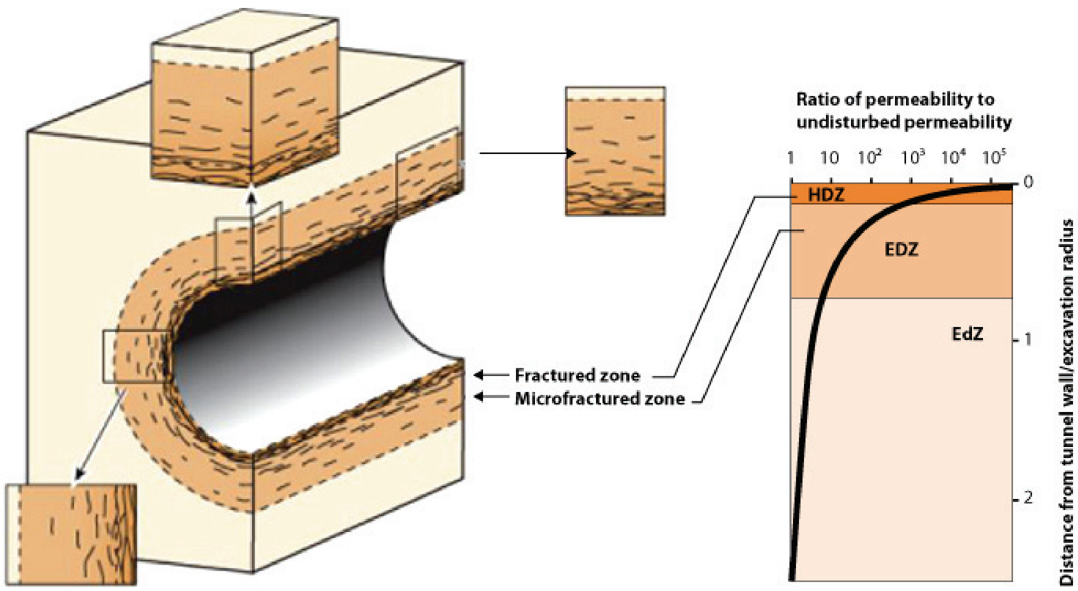
In crystalline rock, such as the Canadian Shield, the presence and distribution of fractures are the most important factors affecting the flow system surrounding an underground repository. The ability to accurately represent the spatial distribution and connections of fractures will increase the confidence of groundwater models, both at local and regional scales. FXSIM3D is research software that was developed for creating 3D fracture network models (FNMs). It is based on a geostatistical procedure that uses a wide variety of different data types regarding the location and orientation of sub-surface fractures. The types of acceptable data for a FNM code are:

- Surface expressions of fractures;
- Statistics on fracture density;
- Structural geology principles that govern down-dip behaviour; and
- Truncation rules for lineament intersection.

The family of FNMs produced by this method are probabilistic in that they consist of equally-likely renditions of fracture geometry. The FNM generated is realistic and structurally complex, honouring the detail of fracture locations, orientations and other aspects of the fracture geometry, such as the down-dip behaviour of fracture surfaces. Such models are useful as input to other types of analyses, from mechanical stability to groundwater flow and contaminant transport modelling, and are also well suited for studying issues involving risk assessment and quantification of uncertainty. Additionally, this method for creating FNMs is useful at all stages of site characterization as it can provide a framework for guiding field investigation activities by reducing the uncertainty in conceptual geosphere models of fractured rock, thereby improving the DGR safety case. In 2010, NWMO continued work towards the main objective of refining a version of the FXSIM3D software to facilitate technology transfer to other users within the geoscience and geological engineering community.

4.2.4.4 Assessment of Excavated Damage Zone

The excavation of any underground opening forms a zone of disturbed rock around it. Within this excavated disturbed zone (EdZ), there exists one or more zones with different degrees of rock damage classified as the excavation damaged zone (EDZ) and the highly damaged zone (HDZ) along the wall of the excavation (Fracture Systems Ltd., 2011). These zones are identified in Figure 4.2 and are collectively referred to as the EDZ. Damage initiation and fracture propagation will occur such that the crack intensity decreases with increasing distance from the excavated wall. The damaged zones in an underground facility, such as the DGR, may create a preferred pathway for radionuclide migration, particularly in a low permeability rock mass.

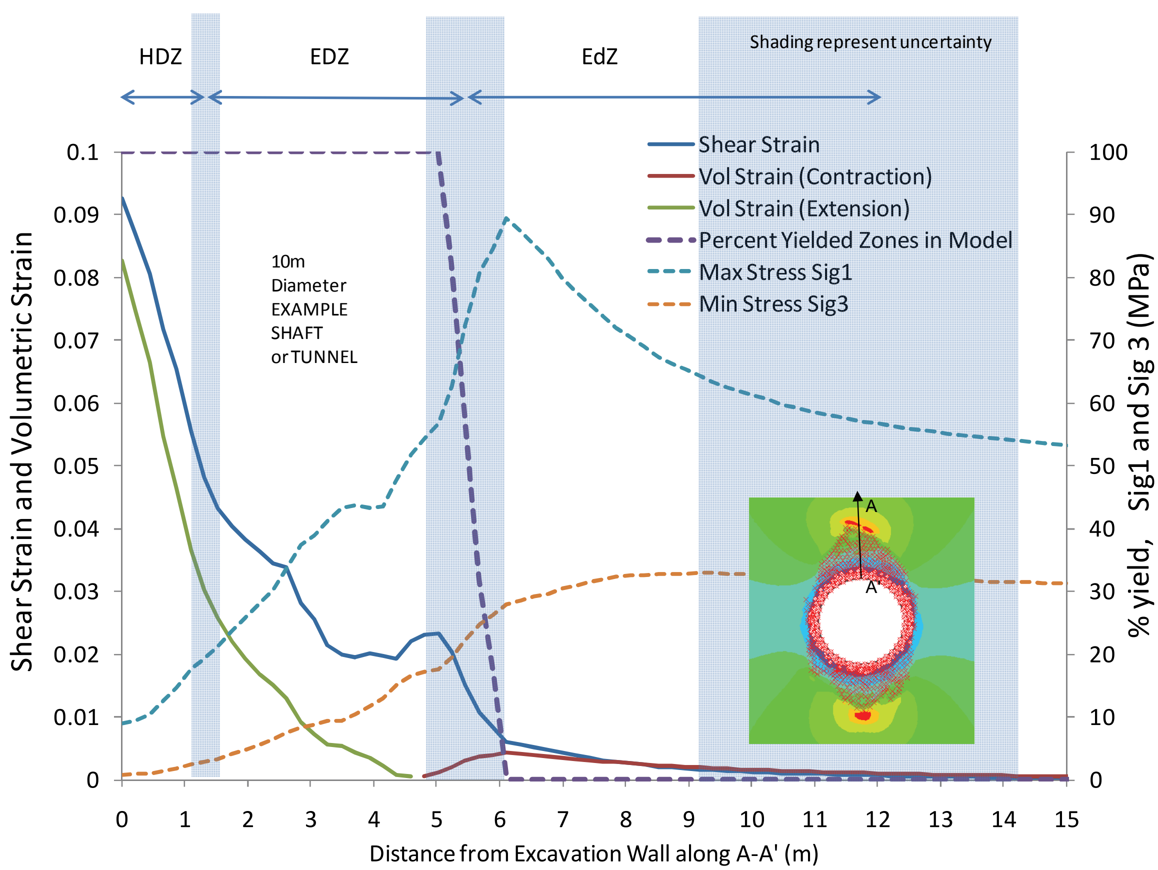


Note: Figure modified after ANDRA (2005)

Figure 4.2: Schematic illustration defining excavated disturbed zone (EdZ), excavated damage zone (EDZ) and highly damaged zone (HDZ) for an unjointed rock

Within the damaged and disturbed zones, state variables such as stress, water pressure, temperature, saturation, water chemistry and related properties, such as porosity, may be altered as a result of excavated openings. However these changes generally do not have a major influence on flow and transport behaviour in the EdZ. Rather, in the EDZ, these same variables can have a long-term effect on porosity and permeability, and hence rock transport properties. Macro-scale fracturing or spalling may occur in the HDZ forming a preferential flow path. The effective permeability of this zone is dominated by the interconnectedness of the discrete fracture system formed and locally may be orders of magnitude greater than the undisturbed rock mass (Fracture Systems Ltd., 2011). Developing an understanding of EDZ evolution, geometry and properties and their influence on the long-term performance of the DGR is an important aspect in the assessment of safety performance and seal cut-off design.

In the APM program, the strategy for assessing the role of the EDZ in the DGR concept is to: (1) Gain an understanding of the role of the EDZ in the possible creation of discrete pathways along the excavated and backfilled openings for mass transport in rock; (2) Minimize damage extent through appropriate excavation methods and geometry of excavated openings based on the knowledge of stress state, re-distribution and influence on EDZ occurrence; and (3) Evaluate and develop sealing methods. A literature review was conducted to document the state of knowledge for predicting the extent and seal cut-off material parameters associated with the Excavation Damaged Zone around bored and blasted tunnels in sedimentary rocks, primarily carbonates and clay-shales. The review also investigates current numerical approaches being employed to determine and evaluate the EDZ and assist with determining the required extent of seal cut-off applications. Figure 4.3 illustrates the use of geomechanical parameters to define the various damaged zones, which is typically continuous and gradational, in a numerical model (Perras et al., 2010). The EDZ research will continue in 2011 and will focus on examining EDZ evolution in limestone formations. These studies include numerical modelling of existing deep excavation openings in mines and hydroelectric projects and geomechanical laboratory testing on quarry limestone samples.



Note: Various indicators for EDZ and HDZ using a strain-weakening model (strength reduces after yield) with a stress ratio of 1.7:1 (horizontal stress: vertical stress). Results are for the vertical line A-A' above the roof. Tunnel diameter is 10 m (Perras, Diederichs and Lam, 2010).

Figure 4.3: Example of typical EDZ model output using arbitrary material parameters

4.2.5 Seismicity

The seismic design of a DGR requires an understanding of anticipated ground shaking at repository depth. In 2010, NWMO activities continued to support seismic monitoring programs by the Canadian Hazards Information Service (CHIS) of Natural Resources Canada (NRC), as detailed in Section 4.2.5.1. In addition, NWMO evaluated the attenuation of seismic motions with depth at the Sudbury Neutrino Observatory (Section 4.2.5.2).

4.2.5.1 Seismic monitoring

NWMO continues to gain knowledge on the seismic characteristics of low seismicity regions of the Canadian Shield in collaboration with the CHIS of the NRC monitoring program. The purpose of NWMO participation is to monitor the low levels of background seismicity in the stable cratonic region of the Canadian Shield. The CHIS monitoring network consists of 18 seismograph stations spreading from Pinawa, Manitoba, in the west, to Chalk River, Ontario, in the east. All stations record real-time, continuous, digital data, which are transmitted by satellite to the NRC data laboratory in Ottawa. In 2009, a total of 82 earthquakes ranging from Nutti magnitude (m_N) 0.7 to 3.4 (Figure 4.4) were located in northern Ontario and eastern Manitoba. Within the 2009 monitoring period, the largest events included a m_N 3.4 in Kirkland Lake and a m_N 2.9 in James Bay in Ontario. The magnitude-recurrence curve of the study area was updated to incorporate the 2009 recorded events and is reported in Hayek et al. (2010).

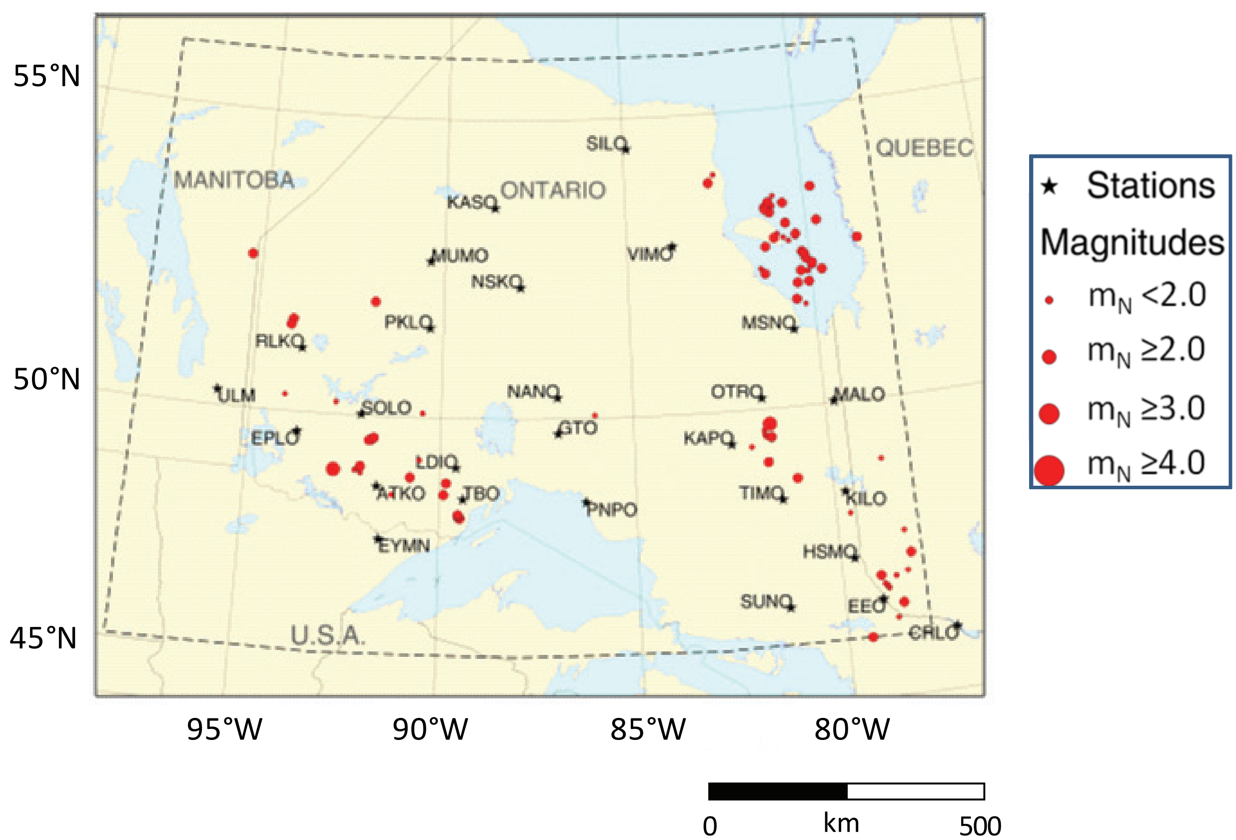


Figure 4.4: Magnitudes (M) of 2009 seismic events in Northern Ontario. Dashed line depicts the study area

4.2.5.2 Underground seismic responses at the Sudbury Neutrino Observatory

The University of Western Ontario compiled a dataset of about 100 events recorded from the surface and underground seismograph stations at the Sudbury Neutrino Observatory (SNO) as part of the POLARIS (Portable Observatories for Lithospheric Analysis and Research Investigating Seismicity) Underground Project. Atkinson and Kraeva (2010) analyzed the collected data and characterized the relationship of surface to underground ground motions, in both the time and frequency domains.

4.2.6 Hydrogeological Modelling Methods

A modelling team from the Université Laval is participating in “Task 7” of the Äspö Task Force on Modelling of Groundwater Flow and Transport of Solutes, which involves the numerical modelling of hydraulic responses in the fractured crystalline rock environment located on Olkiluoto Island in Finland.

The model scale for Task 7 decreased from the large scale of Subtask 7A (about 15 km^3), to the rock-block scale of Subtask 7B (0.032 km^3), and finally to the single-fracture scale of Subtask 7C (36000 m^3). Subtask 7C is the third and final subtask of Task 7, which focuses on simulating fluid flow in low-transmissivity fractures identified during the drilling of ONKPP boreholes and the subsequent construction of the three ONKALO shafts (KU1, KU2 and KU3). The specific goals of 7C are to: (1) Advance the understanding of the role of fracture micro-structural models in performance assessment; 2) Use POSIVA Flow Log (PFL) results to characterize in-plane fracture heterogeneities; 3) Improve the methodology to predict inflow to canister deposition holes; and 4) Assess how data from pilot boreholes can be used to predict flow into canister deposition holes.

The region of interest for 7C covers an area of 30 m x 30 m around each shaft and extends vertically for 40 m. Based on the borehole core analyses, it was assumed that a low-transmissivity fracture crosses the entire model domain. PFL measurements conducted in the ONKPP boreholes were used to calibrate the flow model. The top of these boreholes is at a depth of about 180 m below the floor of the rock caverns along the main tunnel at Onkalo. The length of the boreholes is about 100 m and their diameter is 6 cm. They were drilled before the excavation of the shafts, which had diameters equal to 3.5 m. The boreholes, if left open, would act as conduits for water flow into the rock caverns. Measurements of water leakage rates from a fracture (designated fracture 1) into shaft KU2 were used to test the model.

Due to variability in the magnitude of the measured flow rates at this small scale, the aperture distribution of fracture 1 was assumed to be heterogeneous. The aperture was thus modelled with the random field generator code FGEN (Robin et al., 1993), which can be linked with the numerical simulator HydroGeoSphere (Therrien et al., 2009). Local aperture values were calculated from nine available PFL flow measurements using the cubic law. The aperture mean and variance were then calculated and used as input for the generation of the aperture field for HydroGeoSphere flow and transport simulations. A resulting aperture field is shown in Figure 4.5, where velocity vectors show the inflow distribution to the shaft.

Field observations and modelling results indicate that, for a given shaft, the total flow rate into the shaft is smaller than the PFL flow rates measured at the intersection between fracture 1 and the pilot boreholes prior to drilling the shaft. This difference needs to be further investigated and understood, as it has implications on the approach to evaluating inflow into canister deposition holes. A final report integrating the multi-scale modelling results for Subtasks 7A, 7B and 7C is in preparation and will be finalized during 2011.

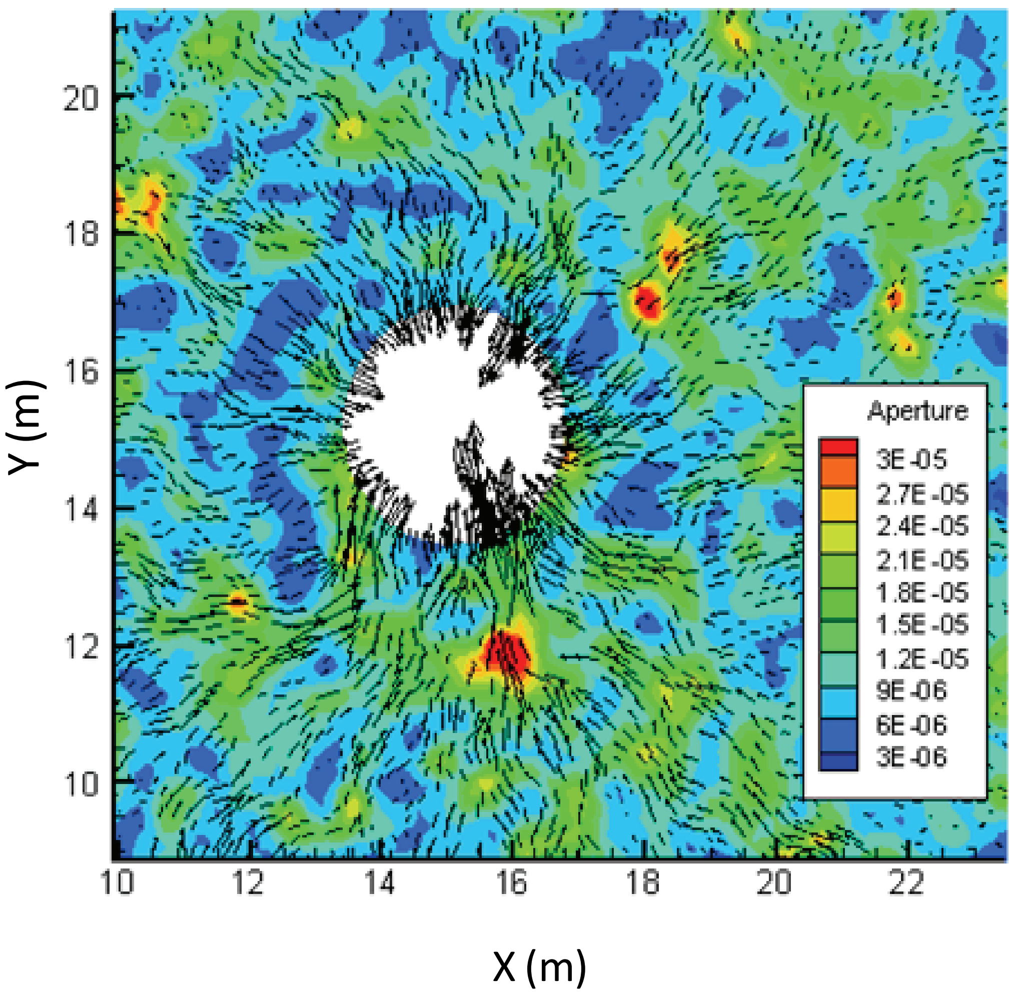


Figure 4.5: Close-up of fracture 1 showing inflow into the shaft. The vector length is proportional to the velocity magnitude (aperture scale is in metres)

4.3 LONG-TERM GEOSPHERE STABILITY

NWMO continues to have an active technical program examining the long-term stability of the geosphere and its resilience to external perturbations in both crystalline and sedimentary settings. Multidisciplinary activities are being directed to improve the understanding of glaciation (Section 4.3.1) and deep groundwater system evolution (Section 4.3.2).

4.3.1 Glaciation

NWMO is refining numerical tools to simulate the impact of long-term climate change on the performance and safety case of a DGR. Potential impacts of glacial cycles on a DGR include: increased stress at repository depth caused by glacial loading; penetration of permafrost to repository depth; recharge of oxygenated glacial melt water to repository depth; and the generation of seismic events and faulting induced by glacial rebound following ice-sheet retreat.

4.3.1.1 Glacial Systems Model

One essential element in assessing the potential impact of glaciation on a DGR is the ability to adequately predict the surface boundary conditions during glaciation cycles. These boundary conditions include permafrost extent and depth, ice-sheet properties (extent, thickness and kinematics), ice-sheet hydrology, as well as other attributes. For NWMO's glaciation case studies, these boundary conditions have been defined based on the University of Toronto's Glacial Systems Model (GSM) predictions (Peltier, 2006). The GSM is a state-of-the-art model of continental scale ice-sheet evolution that has been enhanced to enable calibration using a Bayesian methodology. This methodology allows the model to reconcile a large body of observational constraints governing ice advances and retreats over the North American continent during the Late Quaternary Period of Earth history.

There are several aspects of the GSM that require further development to improve predictions of glaciation boundary conditions and consequently improve the DGR safety case. In the present model, which is based upon the "shallow ice approximation", longitudinal stresses are entirely neglected. Furthermore, the representation of the interaction of ice streams with the ocean, into which they often discharge, is inadequately treated. Since ice streams drain the interior of the ice sheet and thereby control its thickness history, especially insofar as rapid climate change related processes are concerned, the accurate description of such processes is critical to the success of the model. The objectives of the University of Toronto work program are to maintain and improve the GSM. This will be achieved by improving the way that the flow of ice over the landscape is described by implementation of higher degree physics. In addition, the way the model regards climate history will be improved by developing a completely coupled model of the ice-climate system. During 2009-2010, a Discontinuous Galerkin based land-ice model was successfully formulated, implemented and tested.

4.3.1.2 Greenland Analogue Project

In collaboration with SKB and POSIVA, NWMO continued its involvement in the Greenland Analogue Project (GAP) in 2010. The objective of this four-year project (2009-2012) is to advance the understanding of processes associated with glaciation and their impact on the long-term performance of a DGR. The study of the Greenland ice sheet will allow us to increase our understanding of hydrological, hydrogeological and geochemical processes during glacial conditions, and will allow for the refinement of hydrological models used to simulate glacial conditions.

Following an introductory field campaign in 2008 near Kangerlussuaq, Greenland, the GAP field program began in 2009 and included the successful initiation of ice sheet and geosphere/geochemistry studies. Through an extensive field and modelling program, GAP will evaluate glacial hydrology, groundwater flow and groundwater composition (particularly redox conditions) at the base of a continental-scale ice sheet. In 2010, research conducted on the surface of the ice sheet included the installation of GPS stations, ground-based radar work, remote sensing of the study area and tracer tests conducted near the ice margin to look at water flow from the surface to the base of the ice sheet and boreholes were drilled through the ice to the bed. In addition, samples were collected from boreholes drilled in front of the ice sheet to investigate groundwater geochemistry and microbiology (Figure 4.6). A geochemical sampling program of surface water bodies, including a pingo, continued from previous years. In addition, a literature review of hydrogeology/hydrogeochemistry was completed (Wallroth et al., 2010). Preparations for drilling of a deep borehole in 2011 (~500 m) under the margin of the ice sheet were made, including selection of the drilling site.

Figure 4.6: Microbial sample collection for the Greenland Analogue Project

4.3.2 Evolution of Deep Groundwater Systems

NWMO continues to develop numerical methods to assess and quantify the robustness of site characterization data and to predict groundwater migration and transport over geologic timescales as relevant to the safety case. Numerical methods are used to assemble and test descriptive geosphere conceptual models developed through integration of multidisciplinary site characterization data sets. In addition, numerical models are used to refine the understanding of groundwater system evolution in both crystalline Shield rock and sedimentary basin environments.

4.3.2.1 Numerical Groundwater Modelling

Previous work programs (Sykes et al., 2003, 2004 and Normani et al. 2007) have provided insight into the influence of geosphere properties on deep groundwater system dynamics and evolution. Geosphere parameters and properties, such as salinity, fracture networks and fracture zone properties (such as width and porosity) were investigated. The studies also provided a better understanding of the impact of glaciation on deep groundwater systems in Canadian Shield settings and demonstrated the suitability of Mean Life Expectancy as a performance measure when evaluating geosphere responses and parameter uncertainty.

In 2010, further insight into the importance of total dissolved solids distributions and density versus depth and their impact on groundwater flow evolution was gained. In addition, subgridding methods were refined to include site-scale features within a regional scale numerical model. The subgridding method was verified to ensure numerical accuracy through comparison with a nested model-in-model approach. In addition as a part of GAP (Section 4.3.1.2), a groundwater model was created for the Kangerlussuaq field site. This required the development of hydromechanical coupling methods, described in Section 4.2.4.2 that assume homogeneous loads, no lateral strain, and implementation using the one-dimensional loading efficiency.

4.3.2.2 Reactive Transport Modelling

Reactive transport modelling is an approach for assessing long-term geochemical stability in geological formations relevant to a DGR for used nuclear fuel. MIN3P is a multicomponent reactive transport code that was successfully used to evaluate redox stability in crystalline rocks of the Canadian Shield (Spiessl et al., 2009). In 2007, a state-of-science review of reactive transport modelling in sedimentary rocks recommended further development of MIN3P to improve its capabilities for modelling sedimentary rock systems (Mayer and MacQuarrie, 2007). On the basis of the recommendations, a two year research collaboration with University of British Columbia and University of New Brunswick was initiated in 2009 to expand MIN3P applications to sedimentary systems. Mayer and MacQuarrie (2010) summarize the governing equations of MIN3P and evaluate its sensitivity to spatial and temporal discretization parameters.

In 2009, MIN3P was further developed to include density dependent reactive transport (MIN3P-DENS). This was further refined to include the Harvie-Moller-Weare model, which is based on the Pitzer equations for the calculation of activity corrections in high ionic strength solutions. This version was named MIN3P-NWMO and its capabilities and limitations for simulating regional scale reactive transport in sedimentary basins were evaluated.

In 2010, the capabilities of MIN3P-NWMO were further expanded to include: (1) One-dimensional vertical stress to account for the mechanical deformation induced by ice sheet loading, and (2) Energy transport coupled with density dependent flow and reactive transport. The enhanced MIN3P-NWMO code was used to conduct reactive transport simulations in a sedimentary basin affected by a glaciation/deglaciation event so that the evolution of groundwater flow, groundwater chemistry and geochemical processes during the density-driven flow and reactive transport could be evaluated. In addition, a series of alternative scenarios were simulated to investigate the influence of different conceptual models, such as the effects of mineral dissolution-precipitation reactions, permafrost formation in front of the ice sheet, groundwater mixing, cation-exchange reactions, freshwater thickness, thermal convection, and porosity variation with depth on density-driven flow and reactive transport.

In 2011, the MIN3P-NWMO reactive transport code will be applied to simulate in situ disturbances, for example redox and pH changes, in the Disturbances, Diffusion and Retention (DR-A) experiment at the Mont Terri Underground Rock Laboratory. The simulation will use in situ, field-scale data and the results will be compared to the other reactive transport codes.

5. REPOSITORY SAFETY

5.1 ASSESSMENT CONTEXT

The objective of the repository safety program is to evaluate the operational and long-term safety of any candidate deep geological repository (DGR) in order to assess and improve the safety of the proposed facility. In the near-term, before any candidate site has been proposed, the safety objective is addressed through case studies and through improving our understanding of important features and processes.

Garisto et al. (2009) provides a technical summary of information on the safety of a DGR for used fuel. The report summarizes the key aspects of the DGR concept and explains why the repository concept is expected to be safe (see, for example, Table 5.1). The report is non-site specific; it considers alternative geologic settings, specifically both the Canadian Shield and sedimentary rock formations; and encompasses several design concepts.

Table 5.1: Typical Physical Attributes Relevant To Long-Term Safety

- Repository depth provides isolation from human activities
- Site low in natural resources
- Durable waste form
- Robust container
- Clay seals
- Low-permeability host rock
- Spatial extent and durability of host rock formation
- Stable chemical and hydrological environment

5.2 MODEL AND DATA DEVELOPMENT

The objective of this program is to maintain or improve models and data suitable for supporting safety assessment of potential sites and designs. It is divided into five areas discussed in subsections below: wasteform (Section 5.2.1), repository model (container, buffer/backfill seals, near-field rock, Section 5.2.2), geosphere model (including shaft seals, Section 5.2.3), biosphere model (Section 5.2.4), and integrated system model (Section 5.2.5).

5.2.1 Wasteform Modelling

The first barrier to the release of radionuclides is the used fuel matrix. Even if a container fails, most radionuclides remain trapped within the UO_2 grains and are only released as the fuel itself dissolves. The rate of fuel dissolution is therefore an important parameter for long-term safety.

UO_2 dissolves extremely slowly under reducing conditions similar to those expected in a Canadian DGR. However, in a failed container that has filled with groundwater, used fuel dissolution may be driven by oxidants, particularly hydrogen peroxide (H_2O_2), generated by radiolysis of water. The mechanistic understanding of the radiolytic corrosion of UO_2 is therefore important for long-term predictions of used fuel stability.

Within the last several years, dissolved hydrogen gas (H_2) has been confirmed as a key factor in the corrosion process. Hydrogen is also generated from radiolysis, but much larger amounts are generated as a result of corrosion of the inner steel vessel of the container.

The 2010 experimental program on UO_2 dissolution, which was carried out at the University of Western Ontario, investigated the influence of:

(1) H_2 inhibition of UO_2 corrosion in the presence of H_2O_2 (Broczkowski et al., 2010a,b);
(2) Fuel composition on the reactivity of UO_2 (He and Shoesmith, 2010; He et al., 2010): and
(3) Corrosion product deposits on UO_2 dissolution.

A combination of electrochemical and open circuit corrosion measurements on UO_2 electrodes and surface analytical techniques were used in the investigations (He and Shoesmith, 2010; He et al., 2010; Ofori et al., 2010). The tests were conducted mainly with unirradiated 1.5%, 3% and 6% SIMFUELs, representing CANDU fuel burnups from about 210 to 800 MWh/kgU. SIMFUEL (simulated high-burnup UO_2-based fuel) is made by doping unirradiated natural UO_2 pellets with non-radioactive elements to replicate the chemical composition of used fuel, including formation of so-called ε-particles – alloys of the fission products Mo, Ru, Tc, Pd and Rh. The results of this University of Western Ontario research are summarized here in Sections 5.2.1.1 to 5.2.1.3.

Generally, the Zircaloy cladding is conservatively assumed not to provide protection for the used fuel in a defective container. However, the corrosion rate of Zircaloy determines the rate of release of radionuclides bound within the cladding. A solubility-limited dissolution model is currently used in safety assessment models to calculate the dissolution rate of the Zircaloy. In 2010, a literature search was conducted to compile measured corrosion rates of zirconium under repository conditions (Shoesmith and Zagidulin, 2010).

5.2.1.1 Hydrogen Inhibition

In previous years, a series of corrosion potential measurements followed by X-ray photoelectron spectroscopy (XPS) examinations of the surface showed that H_2O_2 and H_2 react synergistically on SIMFUEL electrodes containing epsilon particles. In H_2O_2-containing solutions, UO_2 is oxidized by the reaction of H_2O_2 on both the UO_2 surface and the noble metal particles. These reactions proceed via the formation of OH• surface radicals. The OH• radicals formed on the noble metal particles can be scavenged by reaction with dissolved H_2 leading to a suppression of UO_2 oxidation. Since H_2 concentrations are generally many orders of magnitude greater than H_2O_2 concentrations, the rate of production of surface H• radicals on the noble metal particles is considerably larger than the rate for OH• formation. A $[H_2]/[H_2O_2]$ ratio of $\geq 10^6$ was found to be sufficient to completely protect the UO_2 surface from oxidation (Broczkowski et al., 2010a).

In 2010, a literature review on nuclear fuel dissolution and radionuclide release studies in aqueous solutions containing dissolved hydrogen was completed (Broczkowski et al., 2010b). Studies include investigations with spent pressurized water reactor and mixed oxide fuels, fuel specimens doped with alpha emitters to mimic “aged” fuels, SIMFUELs and unirradiated uranium dioxide pellets and powders. In all these studies, dissolved hydrogen was shown to suppress fuel corrosion and in spent fuel studies to suppress radionuclide release. A number of mechanisms have been proposed to explain these effects, all of which involve the activation of hydrogen to produce the strongly reducing H• radical, which scavenges radiolytic oxidants and suppresses fuel oxidation and dissolution. Both gamma and alpha radiation have been shown to produce H• surface species. In the absence of radiation fields, activation can occur on the surface of noble metal (epsilon) particles. Since these epsilon particles are galvanically-coupled to the fuel matrix they act as anodes for hydrogen oxidation (which proceeds through surface H• species) and forces the UO_2 to adopt a low potential. Depending on the radiation fields present and the number density of epsilon particles, complete suppression of fuel corrosion appears possible even for hydrogen pressures as low as 0.01 to 0.1 MPa. Since the corrosion of steel liners within failed waste containers could produce hydrogen pressures up to 5 MPa, fuel corrosion could be completely suppressed under the long-term conditions expected in sealed repositories.

5.2.1.2 Influence of Degree of Fuel Non-Stoichiometry on Fuel Reactivity

Non-stoichiometry (x in UO_{2+x}) could exist at grain boundaries in the fuel, where lattice defects concentrate. Older fuels may have a higher degree of non-stoichiometry than currently manufactured fuels because formerly, fuel sintering was not as effective as it is today. Also, non-stoichiometry could be produced by vapour phase corrosion in a failed container prior to aqueous immersion. Non-stoichiometry affects both the cathodic (H_2O_2 reduction) and anodic (UO_2 oxidation and dissolution) reactions and therefore it is important to understand the effects of non-stoichiometry on fuel kinetics.

In 2010, examination of a series of well characterized non-stoichiometric UO_{2+x} samples continued. A combination of techniques was used: scanning electron microscopy coupled with energy dispersive X-ray analysis (SEM/EDX), Raman spot analysis (He and Shoesmith, 2010) and scanning electrochemical microscopy (SECM). In these experiments, SECM is used to measure probe approach curves, which are corrected for diffusion of the oxidant from the bulk solution using a finite element model. In this way, the corrosion rates at individual locations could be determined, on a micrometer scale, and the corrosion kinetics could be correlated to surface composition. It was found that stoichiometric locations have extremely low corrosion rates while rates on distorted non-stoichiometric locations were up to 1000 times higher (He et al., 2010).

SECM, coupled with voltametric scanning of the UO_{2+x} substrate, showed that the reactivity at different locations varied as oxidation proceeded. These studies demonstrated that the most catalytic surfaces (i.e., most reversibly oxidizable by O insertion/removal) was achieved for intermediate stoichiometries, an observation consistent with previous measurements showing oxygen reduction was catalyzed on slightly oxidized surfaces (Hocking et al., 1994).

Although the large range of non-stoichiometries studied is not expected to occur in used CANDU fuel, the features controlling the chemical reactivity of the fuel as a function of the degree of non-stoichiometry could be determined. Of key importance is the observation that slightly oxidized locations may be the most likely to corrode because they possess increased anodic reactivity (i.e., UO_2 dissolution), compared to stoichiometric UO_2, and are the most catalytic for the cathodic reduction reaction (i.e., H_2O_2 reduction) required to sustain corrosion. Since non-stoichiometry is most likely to be found in grain boundaries within the fuel, the present conservative assumption that radionuclides at these locations will be instantly released cannot be relaxed.

Studies of the anodic dissolution of SIMFUEL in slightly alkaline (pH = 9.8) sodium carbonate/sodium bicarbonate solutions (with total carbonate concentration between 0.005 to 0.2 M, higher than in typical deep groundwaters in crystalline rock) also show the dependence of fuel reactivity on fuel composition. In these experiments, the mechanism and reaction kinetics (Tafel slopes and reaction orders) of the anodic reaction were elucidated. Although the differences in the reaction kinetics of the anodic reaction, compared to earlier work (Shoesmith et al., 1983; Nicol and Needes, 1977), are not completely understood; there is a strong possibility that they reflect the differences in compositional and physical properties of the UO_2 specimens used in the three experiments (UO_2, SIMFUEL and UO_2/U_4O_9 mixed solid).

5.2.1.3 Influence of Corrosion Product Deposits

In a DGR, the redox conditions at the fuel surface will change with time as the strength of the alpha-radiation field decreases. Thus, oxidation of the fuel surface will likely change the composition of the fuel surface over the repository lifetime. This evolution is expected to cause accumulation of corrosion/dissolution product deposits on the surface of the fuel, which could lead to the following effects: (1) They could block the fuel surface, which would reduce the exposed surface area and suppress the rate of fuel corrosion (Shoesmith, 2000); and (2) They could restrict the diffusive transport of species to (e.g. H_2 from steel corrosion) and from (e.g. radiolytically-produced H_2O_2) the reacting surface. This may lead to localized chemistries within the pores of the deposits on the fuel surface such as the development of localized acidity due to the hydrolysis of dissolved uranium (Shoesmith et al., 2003).

The physical and chemical properties of the deposits formed on the fuel surface are determined primarily by the combination of redox conditions, temperature and groundwater composition (Shoesmith, 2000).

In 2009, corrosion product films on SIMFUEL electrode surfaces were grown galvanostatically. In the galvanostatic method, the electrode is subjected to a particular constant current density and the potential-time transient is measured. Since the reaction rate (i.e., the conversion of UO_2 to UO_2^{2+} in unit time) is proportional to the applied current, the overall reaction rate is fixed. In this manner, the influence of rate on the nature of corrosion product deposits, and the chemical conditions within them, could be studied. Ofori et al. (2010) established a threshold, ~0.25 V (vs. the saturated calomel electrode (SCE)), below which acidification did not occur despite formation of a deposit. For current densities below 1-2 nA/cm^2, the potential approached values well below the threshold potential (i.e., ~ 0.11 V (vs. SCE)). Since the maximum corrosion current densities anticipated under disposal conditions are less than 1 nA/cm^2, the prospects for acidification leading to enhanced corrosion are very remote.

In 2010, progress in this area was limited due to a shortage of SIMFUEL material and difficulties with the electrodes. The shortage of SIMFUEL material is expected to be partially resolved in 2011.

5.2.1.4 Zirconium Corrosion

Zirconium alloys are widely used in nuclear reactors as fuel cladding and as reactor structural elements (i.e., CANDU reactor pressure tubes), and are therefore a component of the waste materials that could be emplaced in a DGR. For this reason, the corrosion mechanisms and rates for relevant zirconium alloys under repository conditions have been reviewed (Shoesmith and Zagidulin 2010). Since titanium and zirconium alloys have many similarities, and because the data base for the corrosion of titanium alloys under repository conditions is considerably more extensive than that for zirconium alloys, the electrochemical and corrosion behaviour of both materials have been compared and evaluated. Although electrochemical studies suggest Zircaloy cladding could be susceptible to pitting, redox conditions within a failed waste container will remain reducing and unable to support this corrosion process. This leaves passive corrosion as the only corrosion mechanism. The available data indicates that the rate of passive corrosion will be very low. A conservative upper limit for the passive corrosion rate of zirconium alloys would be 20 nm/year; however, a value of 5 nm/year is more reasonable. Some studies exist to show rates less than 1 nm/year are likely (Wada et al, 1999).

5.2.1.5 Instant Release Fractions

The “instant release fraction” is an important parameter in the assessment of radiological consequences for scenarios in which water is assumed to enter the container. While the uranium fuel matrix has a very slow dissolution rate, some of the radionuclides present in the fuel sheath gap and fuel grain boundaries are soluble and therefore available for “instant” release in the presence of water.

The instant release fraction for a specific radionuclide is defined as that fraction of the associated total inventory that can rapidly escape from the fuel. Instant release fractions have been measured at AECL Whiteshell Laboratories (Stroes-Gascoyne, 1996); however, these measurements are very costly and therefore difficult to reproduce. Reviews of the international data for instant release fractions have been carried out but limited information is available that is relevant to CANDU fuel.

In late 2010 a work program was initiated to determine if it is possible to compute conservative yet realistic estimates of the instant release fractions. This will be done using a combination of ORIGEN-S and FEMAXI computer codes.

The planned approach is to first compare computed values with experimental data available in Stroes-Gascoyne (1996) to confirm code accuracy. If acceptable results are obtained, the method will then be used to compute instant release fractions for a variety of elements for a range of burnups and average power.

5.2.2 Repository Modelling

The repository or "near-field" region includes the waste, the container, the surrounding seals and other engineered barriers, and the adjacent host rock. Almost all the radioactivity is expected to be isolated and contained within this area. Repository safety work in this near-field region is aimed at improving understanding of the transport-limiting processes around a failed container. Technical work being conducted on container corrosion is described in Section 3.3.

5.2.2.1 Failed Container Model

The used fuel container is also a primary barrier to release of radionuclides. Initially it provides a barrier by preventing any access of water to the used fuel. Eventually the containers will corrode or fail, and water will be able to enter and contact the used fuel. At this time, residual radioactivity within the used fuel may be released into this water due to a combination of “instant release” (Section 5.2.1.5) and extremely slow dissolution (Section 5.2.1).

In 2009, a project was initiated to quantify the effect of corrosion products and varying defect size on the release of radionuclides from both the container/buffer and buffer/host rock boundary. The role of corrosion products is complex. Corrosion products reduce void space in the container and provide additional sorption sites for radionuclides. On the other hand, corrosion products are also much less dense than the carbon steel and exert pressure on the copper container, which could lead to expansion of the defect. Nevertheless, the container copper shell and steel insert would still be present in some form and therefore would still represent a physical constraint on both the rate of water access and the rate of radionuclide release.

The project modelled several container failure geometries designed to encompass a range of likely defect scenarios, both with and without the effect of corrosion products. In this modelling each failure scenario is treated as separate steady-state model. The model geometries included: (i) a pinhole defect of 1 mm diameter (that enlarges subsequently by build-up of the corrosion products); (ii) a 10 mm hole; (iii) a 100 mm hole; (iv) a cylindrical crack that runs around the diameter of the copper canister (10 mm wide); and (v) the complete failure of the canister (i.e. no copper remaining). The releases from the various failure scenarios follow expectations, in that the larger the defect, the larger the release.

After the defect modelling was completed in early 2010, an in-house modelling project was initiated in which defects that vary with time are being modelled using COMSOL. Preliminary results from the in-house modelling matched previous safety assessments results quite well. Additional model features will be added in 2011 and documentation will follow.

5.2.2.2 Radionuclide Solubility

The maximum concentration of radionuclides is limited by their solubility in water. Many potentially important radionuclides, such as plutonium, have very low solubilities under conditions expected around a deep repository and will therefore never mobilize in large amounts.

Solubilities are generally calculated using thermodynamic models, which incorporate data for radionuclide elements, as well as for the water composition and key minerals. There are a number of widely used thermodynamic datasets that support these models, and there is ongoing international work to improve the data.

Throughout 2009 and 2010, NWMO updated the solubility limits of several important radionuclides used in the safety assessment models through a series of work programs. Since there is no unambiguous thermodynamic method to calculate solubilities under highly saline conditions, both the Pitzer (Specific-ion Interaction) and SIT (Specific Ion Theory) approaches were used. The Pitzer dataset was the Yucca Mountain Project Dataset converted to PHREEQC format, while the SIT approach used the ThermoChimie v.7b dataset, developed by ANDRA based on published NEA data, also in PHREEQC format.

Over the span of the project the solubility limits of Am, As, Bi, C, Cu, Mo, Nb, Np, Pa, Pb, Pd, Pu, Ra, Se, Sn, Tc, Th, U, and Zr were calculated. The calculations were carried out for reference groundwaters representative of a crystalline and a sedimentary rock environment (see also Section Geosphere Modelling 5.2.3). The reference crystalline groundwater (CR-10) has approximately 10 g/L of total dissolved solids while the much more saline reference sedimentary groundwater (SR-270) has approximately 270 g/L total dissolved solids. The effect of the near field materials on the groundwater compositions is considered by equilibrating the reference groundwaters with minerals present at repository depth (CR-10 eq and SR-270 eq) and equilibrating with the bentonite buffer and the container (CR-10 NF and SR-270 NF).

The results of all the solubility work can be found in the final report (Duro et al. 2010). A sampling of the calculated solubility limits are presented in Table 5.2 for the CR-10 (eq) and CR-10 NF groundwaters.

Table 5.2: Calculated Solubility Limits and Main Associated Uncertainties for Several Elements Using the ThermoChimie v.7b/SIT Database (TC) and the Yucca Mountain Project/Pitzer Databases (YMP)

Element	GW	Solid phase	Solubility TC	Solubility YMP	Mainly sensitive to	Main uncertainty
Am	CR-10 eq	$AmPO_4{\cdot}xH_2O$ (am) $AmOHCO_3$(s)	$5.14{\cdot}10^{-12}$/*$8.2{\cdot}10^{-12}$ $2.23{\cdot}10^{-5}$	$7.94{\cdot}10^{-12}$ $8.51{\cdot}10^{-6}$	pH, phosphate and carbonate concentration	• solid phase formed: phosphates/carbonates/hydroxides
	CR-10 NF	$AmPO_4{\cdot}xH_2O$ (am) $Am(OH)_3$(s)	$2.63{\cdot}10^{-12}$/*$1.06{\cdot}10^{-9}$ $1.51{\cdot}10^{-6}$	$3.64{\cdot}10^{-11}$ $2.8{\cdot}10^{-7}$		
C	CR-10 eq	$CaCO_3$ (calcite)	$8.29{\cdot}10^{-4}$	$8.98{\cdot}10^{-4}$	pH, Calcium concentration	• Reduction to methane/CO_2
	CR-10 NF		$7.01{\cdot}10^{-5}$	$8.80{\cdot}10^{-5}$		
Cu	CR-10 eq	Cu(s)	$1.36{\cdot}10^{-8}$	Insufficient Data	Eh, chloride concentration	• formation of sulphide
	CR-10 NF		$5.22{\cdot}10^{-15}$			
Mo	CR-10 eq	MoO_2 (s)	$8.72{\cdot}10^{-9}$	Insufficient Data	pH, Eh, Calcium concentration	• solid phase formed; • lack of thermodynamic data in both databases, especially in YMP Database
	CR-10 NF		$3.64{\cdot}10^{-15}$			
Np	CR-10 eq	NpO_2(am)	$1.08{\cdot}10^{-9}$	$1.26{\cdot}10^{-9}$		
	CR-10 NF		$1.00{\cdot}10^{-9}$	$1.21{\cdot}10^{-9}$		
Pu	CR-10 eq	$PuPO_4$ PuO_2(am)	$5.12{\cdot}10^{-12}$/ *$8.24{\cdot}10^{-11}$ $1.50{\cdot}10^{-8}$	$9.27{\cdot}10^{-13}$ $7.01{\cdot}10^{-8}$	Eh, pH, phosphate concentration	• Phosphate concentration: Important influence of the database used.
	CR-10 NF	$PuPO_4$ PuO_2(am)	$8.12{\cdot}10^{-13}$/*$3.2{\cdot}10^{-10}$ $9.10{\cdot}10^{-8}$	$4.26{\cdot}10^{-12}$ $1.44{\cdot}10^{-5}$		
Tc	CR-10 eq	$TcO_2{:}1.6H_2O$	$4.01{\cdot}10^{-9}$	Insufficient Data	Eh	• Lack of thermodynamic data in the YMP Database for aqueous species
	CR-10 NF		$4.05{\cdot}10^{-9}$			
Th	CR-10 eq	ThO_2(aged/am)	$3.94{\cdot}10^{-9}$/*$2.49{\cdot}10^{-8}$	$5.23{\cdot}10^{-9}$ /*$1.68{\cdot}10^{-7}$	Carbonate concentration	• Crystallinity of the solid phase formed
	CR-10 NF		$1.38{\cdot}10^{-9}$/*$7.85{\cdot}10^{-9}$	$3.69{\cdot}10^{-9}$ /*$1.14{\cdot}10^{-7}$		
U	CR-10 eq	UO_2(am)	$3.45{\cdot}10^{-9}$	$4.80{\cdot}10^{-9}$	Eh, carbonate concentration	• Crystallinity of the solid phase formed
	CR-10 NF		$3.16{\cdot}10^{-9}$	$3.82{\cdot}10^{-9}$		

*: Phosphate concentration given by equilibrium with hydroxyapatite.

5.2.2.3 Thermodynamic Database

NWMO continues to support the joint international Nuclear Energy Agency effort on developing thermodynamic databases for elements of importance in the safety assessment of a DGR for used fuel (Mompeán and Wanner 2003). The fourth phase of the Thermochemical Database (TDB) Project began in February 2008 and will continue until 2012. The initial focus is to finalize reviews of chemical thermodynamic data for inorganic compounds and complexes of thorium, iron and tin. In late 2008, the thorium thermodynamic data report was published (Rand et al., 2008). The reviews of tin and the key species of iron (Fe(II) and Fe(III)) are in their final form and they are expected to be published once reviews of other species and compounds of iron are complete.

Due to the high salinity of brines observed in sedimentary and crystalline rocks in Canada, a thermodynamic database including ion interaction parameters is needed for radionuclide solubility calculations. In 2008, the Yucca Mountain Project (YMP) Pitzer database "data0.ypf.R2", which includes Pitzer ion interaction coefficients developed by the Sandia National Laboratories for the Yucca Mountain Project, was converted from EQ3/6 format to PHREEQC format (Benbow et al., 2008). To test the suitability of the database for Canadian conditions, results were compared with other Pitzer thermodynamic databases provided with the standard PHREEQC package and the SIT thermodynamic database ThermoChimie. This led to a new work program to develop a thermodynamic database specifically for Canada.

The purpose of the new program is to extend the Yucca Mountain Project (YMP) Pitzer database “data0.ypf.R2” to provide an improved internally consistent Pitzer database for the Canadian used fuel program. Specific objectives are to: (1) Identify and fill in the data gaps of mineral phases that are important to the crystalline rocks in the Canadian Shield and sedimentary rocks in Canada; (2) Identify and fill in the data gaps of secondary minerals arising from the failure of waste container and dissolution of radionuclides; (3) Identify and fill in the data gaps of aqueous species that are important to the Canadian used fuel program; (4) Evaluate the reliability of the underlying model for redox reactions employed by PHREEQC; (5) Review and add the available relevant Pitzer ion interaction parameters for the aqueous species identified in (1), (2) (3), and (4); and (6) perform internal consistency checking.

5.2.2.4 Gas Transport Through Buffer

Corrosion of steel in the repository will result in the slow generation of gases. The low-permeable clay seal around the container will retain these gases until sufficient pressure is reached, following, which they will escape. The nature of this behaviour is of interest for understanding the behaviour in the near-field around a failed container. To explore this area, a full-scale in situ test called "LASGIT" was initiated several years ago in the SKB Äspö Hard Rock Laboratory in Sweden.

NWMO is contributing to the gas transport modelling component of LASGIT. All gas transport modelling is being conducted for NWMO by Geofirma Engineering. The TOUGH2 two-phase transport code was selected as the reference code and then modified for LASGIT to simulate pressure-induced changes in properties such as micro- and macro- fracturing. In 2006 and 2007, the modified code was applied to laboratory experimental data (MX-80-10 conducted by Harrington and Horseman, 2003) and predictive simulations of the LASGIT experiment.

In 2009, Geofirma Engineering obtained TOUGH2-MP, a parallelized version of the TOUGH2 code. This code has been altered to include LASGIT-specific functions and substantially improve the output algorithm for time dependent data from grid-blocks, connections, and source/sink nodes.

Results of the preliminary gas tests performed in 2008 at LASGIT were analysed in 2009. The results were difficult to interpret, but did seem to indicate that a gas breakthrough had occurred. However, it was concluded that the experimental system might not have been fully saturated and may have been producing unreliable data. The model is currently being updated to include data from a second round of gas injection tests completed in 2010.

5.2.3 Geosphere Modelling

The development of improved geosphere models is largely carried out under the Geoscience work program. Recent safety assessment case studies have used both detailed geosphere models and system-level safety assessment models. In particular, the Third Case Study and Third Case Study/Horizontal Borehole Concept studies (Gierszewski et al., 2004b, Garisto et al., 2005) have used a regional study model similar to that used in on-going Geoscience numerical studies and the FRAC3DVS code (see Section 4.3.2.1) to provide detailed 3-D groundwater flow and transport analyses. This ensures that the same geosphere conceptual model is being used by both the geoscience and safety assessment groups. The Fourth Case Study, initiated in 2009 (see Section 5.3.1.2), considers crystalline rock and is an update of the Third Case Study. The Fifth Case Study will consider a repository located in sedimentary rock and will be initiated in 2011.

Geochemical modelling, sorption experiments, solubility calculations, and safety assessment calculations require reference groundwaters to ensure consistency. For the hypothetical failure of a used fuel container, the solubility of radionuclides released from the container will depend on the chemistry of the groundwater. The sorption of radionuclides onto the engineered barrier materials and the host rock will also depend on the groundwater chemistry, as will potential geochemical reactions. Appropriate selection of reference groundwater compositions is, therefore, essential to obtain sound solubility values.

Reference waters are standardized compositions, which are representative of a certain rock type or a certain set of near-field conditions. NWMO is currently considering both sedimentary and crystalline rocks as potential hosts for a DGR and therefore representative groundwaters for crystalline rock and sedimentary rock were defined (in 2009 and 2010, respectively). The groundwater compositions are based on measurements over many observed groundwater samples, and do not represent a specific site.

This set of representative groundwaters (two examples of which are presented in Table 5.3 for information) are analogous to groundwaters that are likely to be found in potential deep geological used fuel repository sites within Canada. They cover a range of depths, rock types, and salinities and as such assist in current and future modelling activities.

Table 5.3: Illustrative Deep Groundwater Compositions

Groundwater Type	Crystalline Rock CR-10	Sedimentary Rock SR-270-PW
Nominal pH	7	5.8
Environment Type	Reducing	Reducing
Nominal E_H (mV)	-200	-200
Solutes (mg/L)		
Na	1,900	50,100
K	15	12,500
Ca	2,130	32,000
Mg	60	8,200
HCO_3	70	110
SO_4	1,000	440
Cl	6,100	168,500
Br	-	1,700
Sr	25	1,200
Li	-	5
F	2	1
I	-	3
B	-	80
Si	5	4
Fe	1	30
NO_3	<1	<5
PO_4	0	-
Total dissolved salts (mg/L)	11,300	275,000

5.2.3.1 COMSOL Near and Far-Field Modelling

Current NWMO safety assessments assume a hypothetical geosphere with a varied surface topology and numerous lakes and rivers across a ~100km^2 site. In accordance with the surface features, a fracture network is present below surface throughout the geosphere. The properties of the geosphere vary with depth, resulting in several distinct geosphere zones.

In previous safety assessments the repository vault and geosphere have been modelled using FRAC3DVS-OPG, a finite element code developed by the University of Waterloo that solves the 3D advective groundwater flow field and nuclide transport equations (geosciences applications are discussed in Section 4.3.2.1). FRAC3DVS-OPG is the current reference code used by the NWMO; however, other modelling options are being explored.

COMSOL is a generic three-dimensional finite element modelling software package that allows users to represent various geometries and solve coupled equations. COMSOL is currently developed by COMSOL Inc. and has extensive development and support groups as well as a significant international user base. The code has numerous built in equations represented in "modules" as well as the ability for users to specify their own equations or modify existing equations. COMSOL has a module, the earth science module, devoted to modelling flow and transport in both saturated and partially saturated systems.

A work program was initiated in 2010, which involves using COMSOL to model the geosphere around a hypothetical used fuel repository. COMSOL will model the groundwater flow field in the near and far fields (including the fracture network) and the transport of I-129 and U-238 through the model domain and the results will be compared to similar results obtained using FRAC3DVS-OPG. When completed in 2011, this study will provide information on the ease of modelling complex geospheres with COMSOL as well as providing useful validation results for the FRAC3DVS-OPG code.

5.2.4 Biosphere Modelling

5.2.4.1 Iodine in the Biosphere

Iodine-129 is an important radionuclide with respect to a potential long-term public dose impact. In 2002, a literature review was completed and key biosphere model parameters were updated for iodine (Sheppard et al., 2002; 2006). The review indicated several areas where further data would be useful.

One reason for the limited database is that it has historically been difficult to measure iodine in the biosphere because of the low sensitivity of standard analysis procedures. In 2006, a technique was demonstrated that allowed natural iodine levels to be measured using relatively standard equipment. Using natural iodine as an analogue, this opened up an opportunity to improve the I-129 dataset by looking at the natural distribution of iodine.

In 2007, this new approach was used to measure key transfer factor data and obtain other ancillary media parameters in the aquatic and terrestrial ecosystems, which are of interest for safety assessment case studies (e.g. fish, wild game, berries). In 2008, the analysis was extended to include farm environments and domestic animals, as well as a small sampling of tundra ecosystem biota (relevant to periglacial conditions).

In 2009, the three year study and sampling campaign was completed. Sampling focused on aquatic and terrestrial ecosystems as well as domestic farm environments. Key transfer factor data and other ancillary media parameters were measured for a number of biota of interest to safety assessment case studies (e.g. fish, cow, chicken, wild game, berries).

Over the three year study, areas within a representative portion of the Canadian Shield were subdivided into distinct sampling zones to ensure the survey represented the physiographical variation within the larger sampling area. Results show good agreement between measured transfer factor results and plant/soil concentration ratios for iodine compared to those of Sheppard et al. (2002). In many cases, substantial improvements were made to the number of measurements and distribution of a particular transfer factor or concentration ratio when compared with the literature review (see Figure 5.1).

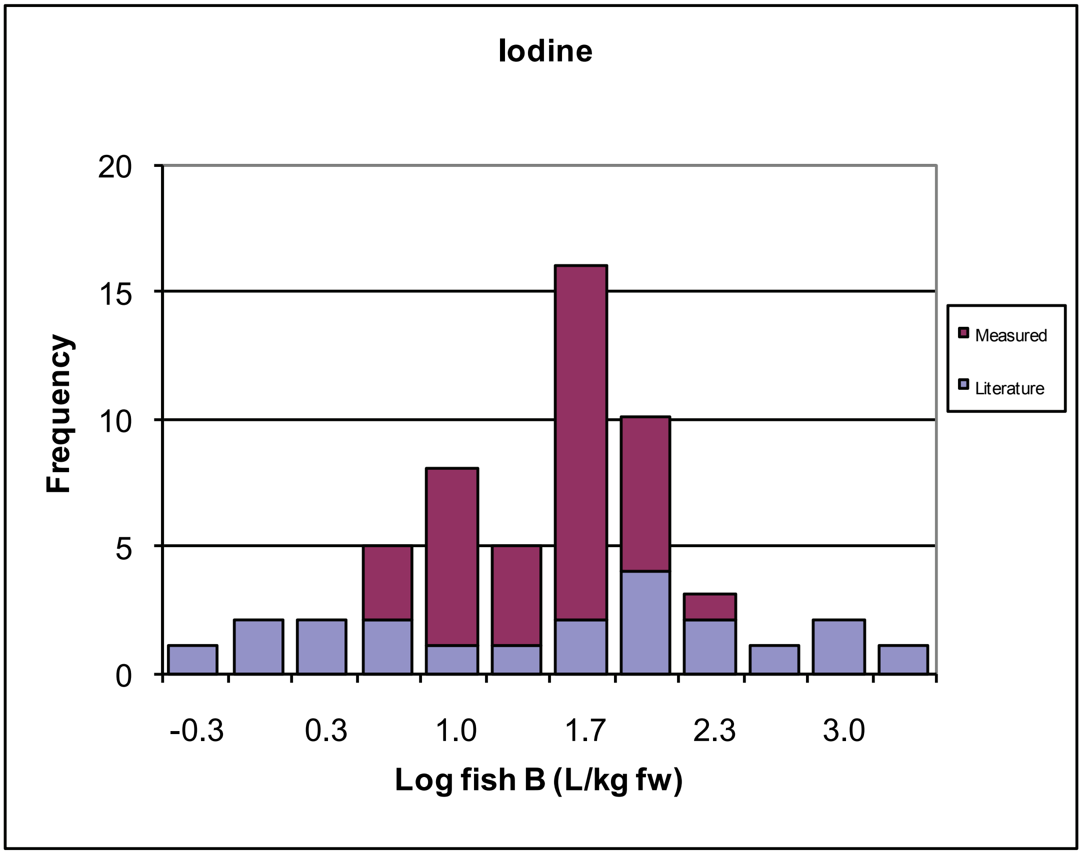


Figure 5.1: Iodine transfer factor data from the literature (blue) compared to additional transfer factors. Measured by Sheppard et al. (2009) from 2007-2009

A cumulative report (Sheppard et al., 2009) presenting all the results of the three year study was published in 2009. In addition, this research resulted in the publication of three peer-reviewed articles in the Journal of Environmental Activity (Sheppard et al 2010a, Sheppard et al 2010b, Sheppard et al 2010c) and study results were presented at the 2010 BIOPROTA meeting in Sweden.

5.2.4.2 Environmental Radioactivity

Postclosure safety assessments use environmental concentrations and fluxes of radionuclides as long-term safety indicators. Knowledge of background radioactivity is useful as a criterion or reference point for these indicators, especially if the data are regionally appropriate. Environmental radioactivity measurements were done in 2010 to complement a literature review completed in 2009 on the summary of background concentrations of radionuclides in surface water and soil across Canada.

Three types of radionuclides were considered. The first radionuclides were primordial, including parents and progeny of ^{235}U, ^{238}U and ^{232}Th, ^{40}K and ^{97}Rb. The second were rare but naturally occurring radionuclides of special interest, including ^{3}H, ^{14}C, ^{36}Cl and ^{129}I. The third were fallout radionuclides with emphasis on ^{3}H, ^{14}C, ^{137}Cs and ^{90}Sr. Data were obtained specifically for Canadian sites, however, data from international sources were also included as needed.

Gaps in information were identified in the 2009 review. In particular, there were few data for ^{129}I and ^{36}Cl. Additionally, the degree of disequilibria in U and Th decay series was not well characterised from the literature. These gaps were addressed by taking background radioactivity measurements of 21 surface waters from New Brunswick to Saskatchewan. Analysis of the samples for ^{129}I and ^{36}Cl was by Accelerator Mass Spectroscopy and for U and Th progeny by radiochemical methods. Trace elements and tritium were also measured. The observed concentrations of ^{129}I and ^{3}H were consistent with the previous review, and the present study increased the numbers of data points several fold. Concentrations of ^{232}Th, ^{235}U and ^{238}U were detectable in all locations, with substantial ranges. The U and Th decay series radionuclides were seldom detectable. ^{36}Cl data are still pending, and will be presented in 2011. The 2009 and 2010 data will be published in one comprehensive report in 2011.

5.2.4.3 No-Effect Concentrations (NECs)

In 2008, a study was performed to present and implement a screening methodology for assessing the potential post-closure impact of a DGR for used fuel on non-human biota. The screening was designed for hypothetical sites representative of selected Canadian conditions under both present and potential future climate conditions. The screening was carried out by comparing estimated radionuclide concentrations to derived “No Effect Concentrations” (NECs), which are screening or threshold criteria. As long as the NECs are not exceeded, there is confidence that, despite uncertainty in modelled environmental concentrations, there would be no significant ecological effect on non-human biota.

In the event NECs are exceeded in screening calculations, a site-specific Ecological Risk Assessment (ERA) would be required to determine whether this is due to conservatism in the assumptions, lack of sufficient data or potential real impact.

NECs were derived in this study for groundwater, soil, surface water and sediment in three ecosystems that represent a range of Canadian conditions, these being Southern Canadian Deciduous Forest, Boreal Canadian Shield Forest, and Tundra (a potential far-future climate condition during glaciation).

Several indicator species were evaluated for each ecosystem, representing a range of different trophic levels within the ecosystem. The NEC corresponding to the most limiting biota for each radionuclide in a particular environmental medium was used as a concentration screening level.

In 2010, a second study was performed, which updated and expanded the previous methodology. The 2010 study expands the original work (Garisto et al., 2008) to include an additional 33 radionuclides and 17 biota, as well as updated transfer factors from the Canadian Standards Association (CSA, 2008) and the IAEA parameter value handbook (IAEA, 2010). The additional radionuclides include a range of long-lived fission products, activation products and actinides with decay chains that are relevant to the assessment of potential impacts. NECs have been re-calculated to incorporate the influence of these new radionuclides and biota.

The new NECs have also been compared to estimated environmental concentrations from the horizontal borehole concept case study (Garisto et al., 2005). The results indicate that there would be no significant radioecological impact on non-human biota for this case study. The results are anticipated to be published as an NWMO Report in 2011.

5.2.4.4 Participation in BIOPROTA

BIOPROTA is an international collaborative forum, which seeks to address key uncertainties in the assessment of radiation doses in the long-term arising from the release of radionuclides as a result of radioactive waste management practices. Participation is aimed at national authorities and agencies with responsibility for achieving safe and acceptable radioactive waste management.

In 2010, NWMO, together with ANDRA, CIEMAT, Enresa, NDA RWMD, NUMO and SKB, financially supported the U-238 Project. The purpose is to improve confidence in long-term dose assessments for U-238 series radionuclides relevant to a used fuel repository at very long times. The goal is to improve understanding in the disequilibrium in the decay chain within geosphere-biosphere interface zones, radon emanation rates and the effect of environmental change.

To date, two workshops have been conducted. The first workshop concluded that the Project should focus on a set of internationally-recommended parameter values, and that a comparison of existing models should be performed. To this end, a hypothetical scenario was designed for testing and code inter-comparison with the results discussed at the second workshop. A second round of testing and code inter-comparison is currently underway, with the focus on predictions of measured soil contaminants at a decommissioned mine site in Spain. Publication of the final report is expected in 2011.

5.2.5 Integrated System Model

The postclosure safety assessment uses several complementary computer codes (Table 5.4). These are either commercially maintained codes, or codes maintained by NWMO under a software quality assurance system.

Table 5.4: Main Safety Assessment Codes for Postclosure Analyses

Software	Description / Use
SYVAC3-CC4	NWMO reference integrated system model
FRAC3DVS	3D groundwater flow and transport code
TOUGH2	3D two-phase gas and water flow code
AMBER	Generic compartment modelling software
COMSOL	3D multi-physics finite element modelling software
PHREEQC	Geochemical calculations code
MICROSHIELD	Radioactive shielding and dose code

The main software activity in 2010 was the development of a new version of the integrated system model SYVAC3-CC4 v.8 and the associated dataset. The code changes addressed minor clean up items found after release of the previous version, while the dataset had numerous updates for the Fourth Case Study, (e.g., revised geosphere, updated repository design, updated container design, and updated biosphere parameters). Other developments included creation of new versions of the pre- and post-processing programs used to format input data and display output data for SYVAC3-CC4.

5.3 CASE STUDIES

The objective of safety case studies is to provide illustrative examples of repository safety under various conditions and to test or demonstrate NWMO's safety assessment approach.

Three major safety assessment postclosure case studies have been considered within the Canadian program, these being the Environmental Impact Assessment (EIS) study (AECL, 1994), the Second Case Study (SCS) (Goodwin et al., 1996) and the Third Case Study (TCS) (Gierszewski et al., 2004b). These case studies provide an opportunity to assess and illustrate the safety implications of the DGR concept in the Canadian Shield. Additional case study work has since been initiated and progress made in 2010 on the Glaciation Scenario, the Fourth Case Study and the new "Fifth Case Study" is described here.

In addition to the postclosure studies, two preclosure studies were also initiated. One study examines postulated releases during normal and accident conditions to determine the required distance to the exclusion zone boundary and the other study provides information on some aspects of conventional safety. These studies are also described below.

5.3.1 Postclosure Case Studies

5.3.1.1 Glaciation Scenario

The reference time frame for the safety assessment of deep repositories is one million years, roughly equivalent to the time scale for the radioactivity in used fuel to decrease to that due to its natural uranium content. Over the past one million years, the most significant natural event across Canada has been repeated glaciation cycles, which have occurred approximately every 100,000 years. It is possible that current greenhouse gas levels will delay the onset of the next glaciation, but in the long run it is prudent to assume that the glacial cycles will resume because they are driven by long-term variation in solar insolation due to earth's orbital variations.

During past glacial cycles, much of Canada has been covered by kilometre-thick ice sheets. Because these glacial cycles represent such a large potential perturbation to a site, the Canadian used fuel program has been examining the implications of glaciation for many years (see, for example, Section 4.3.1 for recent work in the Geoscience program). The general conclusion is that an appropriately sited and sufficiently deep repository can provide containment and isolation of the used fuel during glaciation.

In 2009, the effects of an evolving climate with multiple glaciations were quantitatively evaluated, from a safety assessment perspective, within the context of the hypothetical Third Case Study site on the Canadian Shield. The purpose of this "Glaciation Scenario" case study was to quantitatively assess the long-term dose implications of glacial cycles for a DGR, and to understand the key factors involved. The results of this study, which were summarized in previous annual reports, were published in 2010 in two peer-reviewed reports: Garisto et al. (2010) and Walsh and Avis (2010). Based on this study, it was concluded that the potential releases from a DGR would remain well below regulatory limits even when the effects of glaciation are considered. The results of the Glaciation Scenario study were also presented to SKB and at the BIOPROTA annual meeting in Stockholm.

5.3.1.2 Fourth Case Study

The Fourth Case Study builds on the series of postclosure safety assessments for a DGR in crystalline rock. Key differences relative to the Third Case Study are the shallower depth (500 m), use of in-floor emplacement, a larger container and a different geosphere. The study determines hypothetical radiological and non-radiological impacts to humans and non-human biota for a range of Normal Evolution and Disruptive Event scenarios. Modelling and assessment are based on information expected to be available during the site evaluation stage, but prior to exploratory drilling.

The used fuel container consists of an outer copper vessel, an inner steel vessel, and three steel baskets as shown in Figure 3.2. The copper vessel provides a corrosion-resistant barrier in the repository environment. The inner vessel is designed to withstand any mechanical stresses, including stresses due to glaciation. Each container holds 360 used fuel bundles, distributed in six layers of 60 bundles each held in three baskets that are stacked on top of each other inside the inner vessel (two layers per basket).

Once placed in the repository, used fuel containers are surrounded by compacted bentonite clay. All excavated spaces are filled with mixtures of clay, sand, and rock to minimise the flow of water. In addition, placement rooms are sealed with bulkheads of special high-performance concrete. Shafts are similarly filled and sealed, isolating the repository from the biosphere.

The geosphere in the Fourth Case Study is composed of granitic rock, characterised by an intermediate permeability and a statistically generated discrete fracture network. This geosphere is considered representative of potential crystalline rock settings, though not necessarily with the best or worst possible features. The geosphere permeability model for the hypothetical site assumes bulk rock permeability decreases with depth, with a permeability of 8.3×10^{-20} m² at repository depth (500 m). Fractures are conservatively assumed to be uniformly, highly permeable, with a permeability of 4.1×10^{-14} m² from repository depth to surface. The watershed area and location of the repository are shown in Figure 5.2.

The FRAC3DVS-OPG code is used to calculate the groundwater flow field and the results are processed to illustrate the route particles originating in the repository would take to surface if they moved with the groundwater (Figure 5.3). These particles reach the surface at the discharge locations shown in Figure 5.4.

In 2010, revisions and updates to the FRAC3DVS-OPG and the SYVAC3-CC4 codes were completed and a number of test runs performed. Detailed analysis of contaminant flows using both deterministic and probabilistic methods will be completed in 2011 with the results to be sent to the CNSC as part of the pre-project review (Section 2.1).

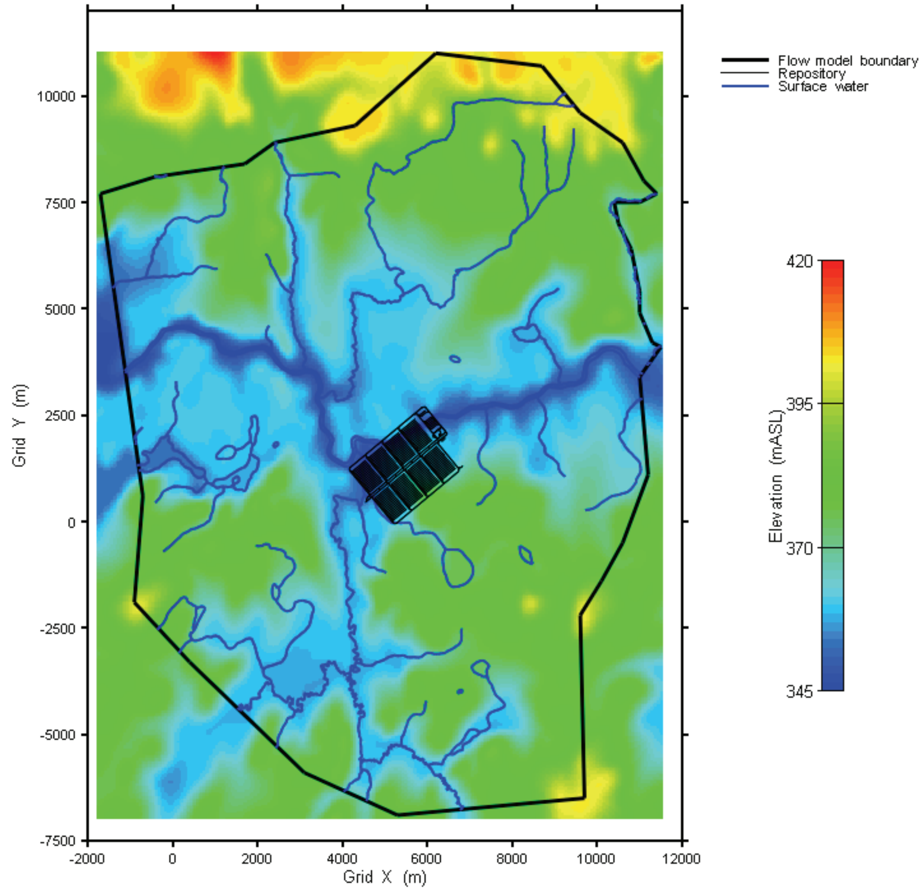


Figure 5.2: Watershed area and the location of the repository

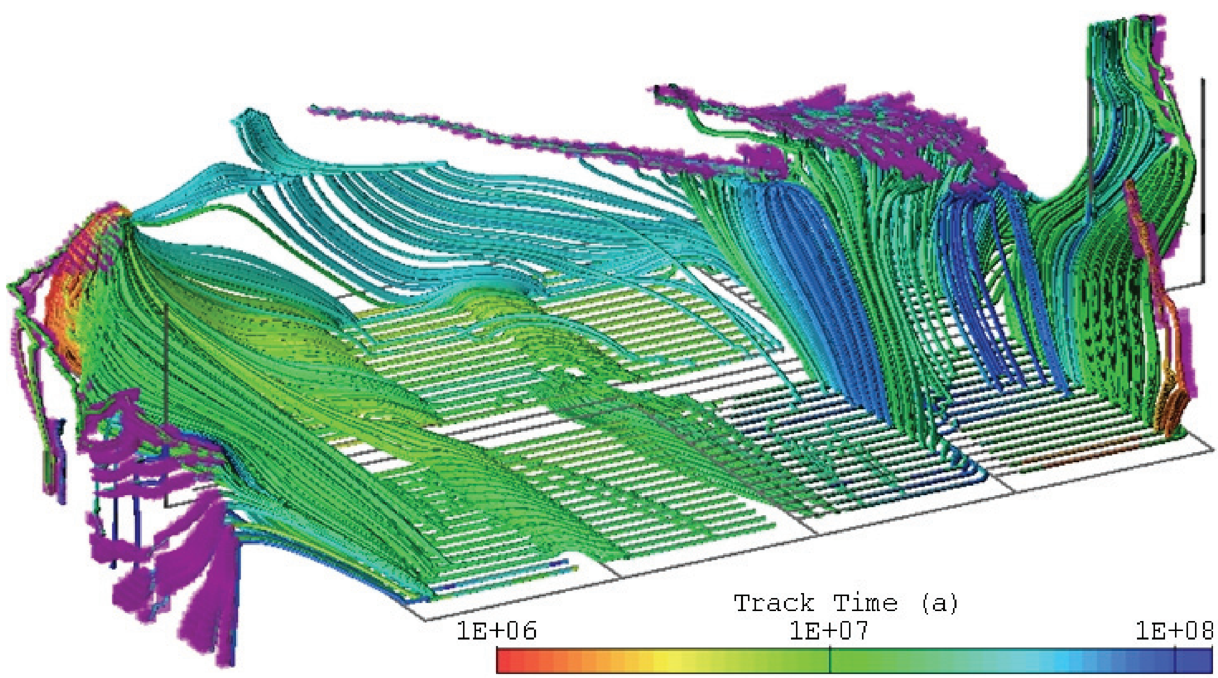


Figure 5.3: Perspective view of particle tracks from repository to surface

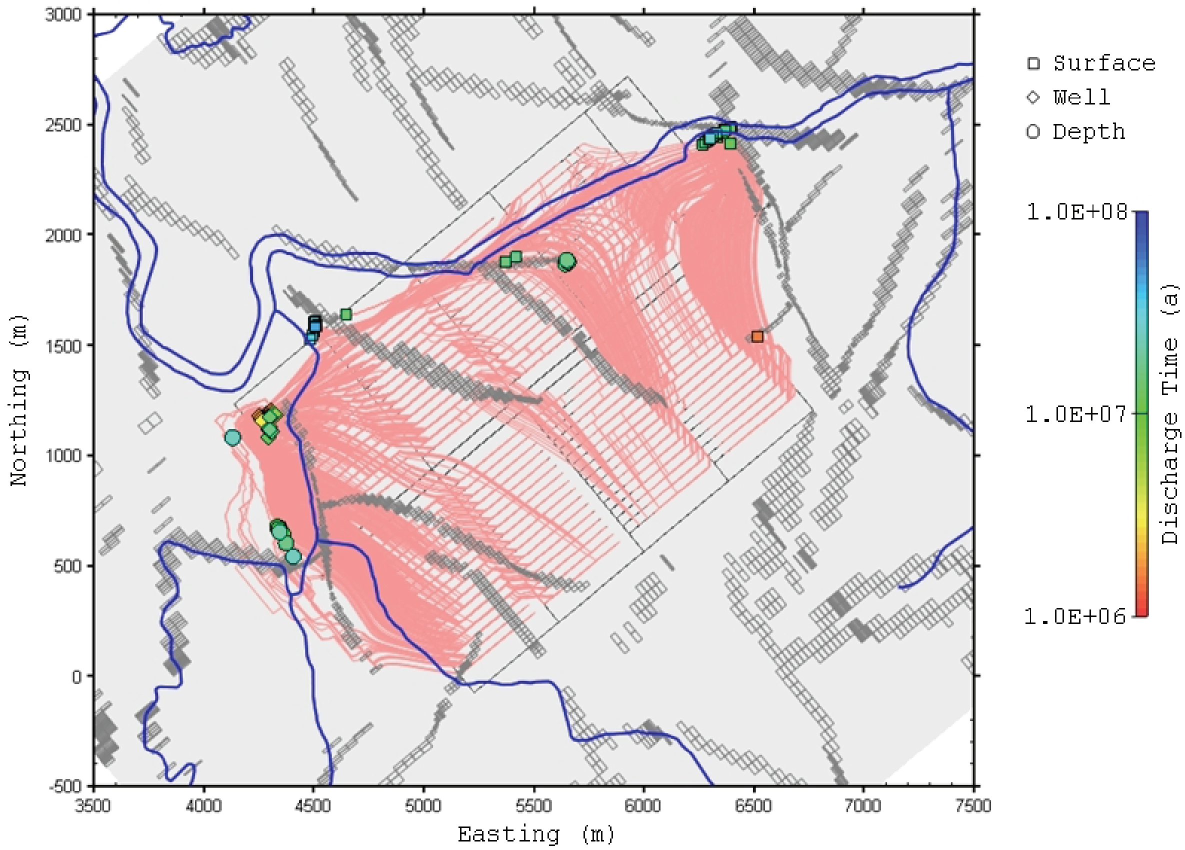


Figure 5.4: Plan view of particle track surface discharge locations

5.3.1.3 Fifth Case Study

A safety assessment for a DGR for used fuel in sedimentary rock will be undertaken in 2011. This case study is nominally named the Fifth Case Study. The repository is assumed to be at a depth of 500 m.

The safety assessment will examine a repository design with horizontal tunnel container placement. In 2010, preparatory work consisting of data compilation, planning, and defining model review and development activities was initiated. The main focus of the 2011 effort will be on corrosion and gas modelling to explore the postclosure safety implications of steel used fuel containers (i.e., no copper shell).

5.3.2 Preclosure Studies

5.3.2.1 Exclusion Zone Boundary

Hypothetical radiological emissions due to operation of a DGR were used to estimate the minimum standoff distance necessary to ensure any consequent dose to members of the public would not exceed a small percentage of the regulatory dose limit. Public dose was calculated at distances ranging from 100 m to 2000 m from the aboveground facilities of the DGR. Estimating public dose required modelling of the cumulative public dose from hypothetical exposure to radionuclides entering the environment, as well as from hypothetical exposure to direct radiation.

Environmental pathway analyses were performed as outlined in CSA Standards N288.1 (CSA, 2008) and N288.2 (CSA, 1991). The DGR was considered to have been sited adjacent to a farmed area on the Canadian Shield.

Normal operations could result in emissions of radioactivity, for example from routine used fuel staging and handling, as well as from direct radiation. Preliminary conservative estimates of chronic public dose consequences due to normal operations were orders of magnitude below the regulatory dose limit.

Anticipated Operational Occurrences (AOOs) are considered outside the scope of normal operations, but are expected to occur at least once during every 100 years of operation. Several AOOs were considered in this assessment: (1) An Irradiated Fuel Transportation Cask carrying water from the Irradiated Fuel Bay; (2) Significantly longer transportation or staging times; (3) Increased processing load; and (4) A 5-fold increase in pre-existing fuel sheath failures. Preliminary conservative estimates of the consequent public dose were also orders of magnitude below the regulatory limit.

Design Basis Accidents (DBAs) are expected to occur less frequently than AOOs, but at least once every 10,000 years of operation. The postulated DBAs considered in this assessment were: (1) Scissor lift failure causing an Irradiated Fuel Transfer Cask to fall and spill; and (2) Overhead carriage failure causing one fuel module to fall on another. Beyond Design Basis Accidents (BDBAs) are expected to occur less frequently than DBAs. The postulated BDBA considered in this assessment was shaft hoist failure causing a Used Fuel Container to drop down the DGR shaft. All accident scenarios were considered both with and without concurrent failure of the emergency HEPA filtration system. Draft documentation prepared in 2010 will be issued in 2011.

5.3.2.2 Conventional Safety Assessment

An updated conventional safety assessment of a DGR for used fuel was completed in 2010 (the last conventional safety assessment was conducted in 1994). The updated assessment was based on the design update by CTECH (2002) and describes the high-level conventional safety aspects, including quantitative estimates of the hazards associated with construction, operation and decommissioning of the facility. The results are shown in Table 5.5.

The assessment accounted for the following:

- changes to the former conceptual design;
- new technology and/or procedures used in comparable large construction and mining projects;
- current mine construction and operating safety experience, both within Canadian and international mining industries; and
- changes in applicable Canadian regulations.

Table 5.5: Comparison of 1994 and 2010 Annual Conventional Safety Risk Estimates

	Site Preparation		Construction		Operation		Decommissioning	
	1994	2010	1994	2010	1994	2010	1994	2010
Fatalities per year								
Surface	–	0.2	0.0	0.0	0.2	0.1	0.0	0.0
Underground	–	–	0.0	0.0	0.1	0.1	0.0	0.0
Total	–	0.2	0.1	0.1	0.3	0.2	0.0	0.0
Injuries per year								
Surface	–	18	5.1	1.4	48	6.0	0.1	0.0
Underground	–	–	5.9	4.1	13	9	2.0	1.1
Total	–	18	11.0	5.5	61	15	2.1	1.1

The accuracy of the annual risk estimates is largely dependent on the projected number of hours per year worked performing a specific type of activity and is therefore commensurate with the accuracy of the available labour estimates. The risk estimates will be revised as more accurate labour estimates become available.

The changes in the conceptual design did not significantly change the safety implications. The assessment found very little difference in the fatality risk estimates for the 1994 and 2002 conceptual designs. Injury estimates did show significant reductions in the construction and operation phases and these are attributed to improvements in building methods, equipment, regulation and materials.

There were several challenges encountered in estimating conventional safety. Workplace injury and fatality data are collected by industry-specific safety organizations, workers' compensation boards and various provincial and federal government departments. The sources of this information may be recorded workers' compensation claims, voluntary reporting to industry associations, and regulatory reporting requirements to the various levels of government. The information available from the various organizations is not readily comparable in all cases because it is usually highly aggregated and in many cases not uniformly grouped or categorized by the different collecting agencies. Finally, the data available represents a lagging indicator of safety performance, therefore predictions based on this data should be considered very conservative.

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Vilks, P. and N.H. Miller. 2010. Laboratory Bentonite Erosion Experiments in a Synthetic and a Natural Fracture. Nuclear Waste Management Organization Report NWMO TR-2010-16. Toronto, Ontario.

Villagran, J., M. Ben Belfadhel, K. Birch, J. Freire-Canosa, M. Garamszeghy, F. Garisto, P. Gierszewski, M. Gobien, S. Hirschorn, N. Hunt, M. Jensen, A. Khan, E. Kremer, G. Kwong, T. Lam, P. Maak, J. McKelvie, C. Medri, A. Murchison, S. Russell, M. Sanchez-Rico Castejon, U. Stahmer, E. Sykes, A. Urrutia-Bustos, A. Vorauer, T. Wanne and T. Yang. 2011. RD&D Program 2011 – NWMO's Program for Research, Development and Demonstration for Long-Term Management of Used Nuclear Fuel. Nuclear Waste Management Organization Report NWMO TR-2011-01. Toronto, Canada

Wada, R., T. Nishimura, K. Fujiwara, M. Tanabe and M. Mihara. 1999. Experimental study on hydrogen gas generation rate from corrosion of Zircaloy and stainless steel under anaerobic alkaline conditions. Radioactive Waste Management and Environmental Remediation — ASME American Society for Mechanical Engineering, Fairfield, New Jersey.

Wallroth, T., H. Lokrantz and A. Rimsa. 2010. The Greenland Analoque Project (GAP) Literature review of hydrogeology/hydrogeochemistry. SKB R-10-34.

Walsh, R. and J. Avis. 2010. Glaciation Scenario: Groundwater and radionuclide transport studies. Nuclear Waste Management Organization Technical Report NWMO TR-2010-09. Toronto, Canada.

Wersin, P., S. Stroes-Gascoyne, F.J. Pearson, C. Tournassat, O.X. Leupin and B.Schwyn. 2011. Biogeochemical Processes in a Clay Formation In-situ Experiment: Part G - Key interpretations & conclusions. Implications for repository safety. Applied Geochemistry. In press.

APPENDIX A: TECHNICAL REPORTS, RESEARCH PAPERS, CONTRACTORS AND AWARDED SCHOLARSHIPS

A.1 NWMO Technical Reports

McKelvie, J., M. Ben Belfadhel, K. Birch, J. Freire-Canosa, M. Garamszeghy, F. Garisto, P. Gierszewski, M. Gobien, S. Hirschorn, N. Hunt, A. Khan, E. Kremer, G. Kwong, T. Lam, H. Leung, P. Maak, C. Medri, A. Murchison, S. Russell, M. Sanchez-Rico Castejon, U. Stahmer, E. Sykes, A. Urrutia-Bustos, J. Villagran, A. Vorauer, T. Wanne and T. Yang. 2010. Technical Program for Long-Term Management of Canada's Used Nuclear Fuel – Annual Report 2009. Nuclear Waste Management Organization Report NWMO TR-2010-01. Toronto, Canada

Duro, L., V. Montoya, E. Colàs and D. Garcia. 2010. Groundwater equilibration and radionuclide solubility calculations, Nuclear Waste Management Organization Report NWMO TR-2010-02. Toronto, Canada.

Cavé, L., T. Al, Y. Xiang and D. Loomer. 2010. Investigations of diffusive transport processes in sedimentary rock. Prepared by the University of New Brunswick. Nuclear Waste Management Organization Report NWMO TR-2010-04. Toronto, Canada.

Stroes-Gascoyne, S. C.J. Hamon, D.A. Dixon and D.G. Priyanto. 2010. The Effect of $CaCl_2$ Porewater Salinity (50-100 g/L) on the Culturability of Heterotrophic Aerobic Bacteria in Compacted 100% Bentonite with Dry Densities of 0.8 and 1.3 g/cm^3. Prepared by Atomic Energy of Canada Limited. Nuclear Waste Management Organization Report NWMO TR-2010-06. Toronto, Canada.

Newman, R. C.[1], S. Wang[1] and Kwong G[2]. 2010. Anaerobic Corrosion Studies of Carbon Steel Used Fuel Containers. Prepared by [1]University of Toronto and [2]Nuclear Waste Management Organization. Nuclear Waste Management Organization Report NWMO TR-2010-07. Toronto, Canada

Stroes-Gascoyne, S.and C.J Hamon. 2010. The effects of elevated temperatures on the viability and culturability of bacteria indigenous to Wyoming MX-80 bentonite. Prepared by Atomic Energy of Canada Limited. Nuclear Waste Management Organization Report NWMO TR-2010-08. Toronto, Canada.

Walsh, R. and J. Avis. 2010. Glaciation Scenario: Groundwater and radionuclide transport studies. Prepared by Intera Engineering. Nuclear Waste Management Organization Report NWMO TR-2010-09. Toronto, Canada.

Garisto, F.[1], J. Avis[3], T. Chshyolkova[2], P. Gierszewski[1], M. Gobien[1], C. Kitson[2], T. Melnyk[2], J. Miller[2], R. Walsh[3] and L. Wojciechowski[2]. 2010. Glaciation Scenario: Safety Assessment for a Deep Geological Repository for Used Fuel. Prepared by [1]Nuclear Waste Management Organization, [2]Atomic Energy of Canada Limited and [3]Intera Engineering. Nuclear Waste Management Organization Report NWMO TR-2010-10. Toronto, Canada.

Lampman, T. and A. Popescu. 2010. Initial Evaluation of Mechanical Stress Distributions in Spent CANDU Fuel Bundles. Prepared by AMEC NSS Limited. Nuclear Waste Management Organization Report NWMO TR-2010-11. Toronto, Canada

Popescu, A. and T. Lampman. 2010. CANDU Fuel Element Model Development and Sensitivity Study. Prepared by AMEC NSS Limited. Nuclear Waste Management Organization Report NWMO TR-2010-12. Toronto, Canada.

Lampman, T. and P. Gillespie. 2010. Assessment of Delayed Hydride Cracking in Spent CANDU Fuel Bundles during Dry Storage. Prepared by AMEC NSS Limited. Nuclear Waste Management Organization Report NWMO TR-2010-13. Toronto. Canada

Hayek, S.J, J.A. Drysdale, J. Adams, V. Peci, S. Halchuk and P. Street. 2010. Seismic Activity in the Northern Ontario Portion of the Canadian Shield: Annual Progress Report for the Period January 01 – December 31, 2009. Prepared by Canadian Hazards Information Service. Nuclear Waste Management Organization Report NWMO TR-2010-15 . Toronto, Canada.

Vilks, P. and N.H. Miller. 2010. Laboratory Bentonite Erosion Experiments in a Synthetic and a Natural Fracture. Prepared by Atomic Energy of Canada Limited. Nuclear Waste Management Organization Report NWMO TR-2010-16. Toronto, Ontario.

Garamszeghy, M. 2010. Nuclear Fuel Waste Projections in Canada – 2010 Update. Nuclear Waste Management Organization Technical Report TR-2010-17. Toronto, Canada.

Shoesmith, D.W. and D. Zagidulin. 2010. The Corrosion of Zirconium Under Deep Geological Repository Conditions. Prepared by the University of Western Ontario. Nuclear Waste Management Organization Technical Report TR-2010-19. Toronto, Canada.

Maak, P.[1], K. Birch[1] and G.R. Simmons[2]. 2010. Evaluation of Container Placement Methods for the Conceptual Design of a Deep Geological Repository. Prepared by [1]Nuclear Waste Management Organization and [2]G.R. Simmons & Associates Consulting Services Ltd. Nuclear Waste Management Organization Report NWMO TR-2010-20. Toronto, Canada.

King, F. 2010. Stress Corrosion Cracking of Carbon Steel Used Fuel Containers in a Canadian Deep Geological Repository in Sedimentary Rock. Prepared by Integrity Corrosion Consulting Ltd. Nuclear Waste Management Organization Report NWMO TR-2010-21. Toronto, Canada.

Guo, R. 2010. Coupled thermal-mechanical modelling of a deep geological repository using the horizontal tunnel placement method in sedimentary rock using CODE_BRIGHT. Prepared by the Atomic Energy of Canada Limited. Nuclear Waste Management Organization Report NWMO TR-2010-22. Toronto, Ontario.

Jackson, D. and K. Dormuth. 2010. Watching Brief on Reprocessing, Partitioning and Transmutation (RP&T) and Alternative Waste Management Technology – Annual Report. Prepared by David P. Jackson & Associates Ltd. Nuclear Waste Management Organization Report NWMO TR-2010-24.

Shek, G.K. 2010. Determination of the Threshold Stress Intensity Factor and Velocity of Delayed Hydride Cracking of Endplate Welds in CANDU Fuel Bundles with Different Design and Manufacturers. Prepared by Kinetrics Inc. Nuclear Waste Management Report NWMO TR-2010-25. Toronto, Canada.

Dixon, D. A., J.B. Martino and D.P. Onagi. Enhanced Sealing Project (ESP): Design, Construction and Instrumentation Plan. Prepared by Atomic Energy of Canada Ltd. Nuclear Waste Management Organization Report APM-REP-01601-0001.

Martino J.B. and R. Kaatz. Enhanced Sealing Project (ESP): Pre-Construction Grouting Around Seal Locations. Prepared by Atomic Energy of Canada Ltd. Nuclear Waste Management Organization Report APM-REP-01601-0002.

A.2 Publications and Presentations

Refereed Journals

Atkinson, G.M. and N. Kraeva. 2010. Ground Motions Underground Compared to Those on Surface: A Case Study from Sudbury, Ontario, Bulletin of the Seismological Society of America. 100(3): 1293-1305.

Broczkowski, M.E., P.G. Keech, J.J. Noel and D.W. Shoesmith. 2010. Corrosion of Uranium Dioxide Containing Simulated Fission Products in Dilute Hydrogen Peroxide and Dissolved Hydrogen. Journal of The Electrochemical Society, 157:C275-C281.

He, H. and D.W. Shoesmith. 2010. Raman spectroscopic studies of defect structures and phase transitions in hyperstoichiometric UO_{2+x}. Physical Chemistry Chemical Physics, 12:8108-8117.

He, H., Z.Qin and D.W.Shoesmith. 2010. Characterizing the relationship between hyperstoichiometry, defect structure, and local corrosion kinetics of uranium dioxide. Electrochimica Acta, 56:53-60.

Mayer K.U. and K.T.B. MacQuarrie. 2010. Solution of the MoMas reactive transport benchmark with MIN3P-model formulation and simulation results. Computational Geosciences, 14:405–419.

Ofori, D., P.G. Keech, J.J. Noel and D.W. Shoesmith. 2010. Influence of deposited films on the anodic dissolution of uranium dioxide. Journal of Nuclear Materials, 400:84-93.

Sheppard, S.C., J.M. Long and B. Sanipelli. 2010. Verification of radionuclide transfer factors to domestic-animal food products, using indigenous elements and with emphasis on iodine. Journal of Environmental Radioactivity, 101:895-90.

Sheppard, S.C., J.M. Long and B. Sanipelli. 2010. Measured elemental transfer factors for boreal hunter/gatherer scenarios: fish, game and berries. Journal of Environmental Radioactivity, 101:902-909.

Sheppard, S.C., J.M. Long and B. Sanipelli. 2010. Plant/soil concentration ratios for paired field and garden crops, with emphasis on iodine and the role of soil adhesion. Journal of Environmental Radioactivity, 101:1032-1037.

Stroes-Gascoyne, S. 2010. Microbial occurrence in bentonite-based buffer, backfill and sealing materials from large-scale experiments at AECL’s Underground Research Laboratory. Applied Clay Science, 47:36-42.

Stroes-Gascoyne, S., C.J. Hamon, P. Maak and S. Russell. 2010. The effects of the physical properties of highly compacted smectitic clay (bentonite) on the culturability of indigenous microorganisms. Applied Clay Science, 47:155-162.

Stroes-Gascoyne, S., C.J. Hamon and P. Maak. 2011. Limits to the use of highly compacted bentonite as a deterrent for microbiologically influenced corrosion in a nuclear fuel waste repository. Physics and Chemistry of the Earth. Accepted.

Stroes-Gascoyne, S., C. Sergeant, A. Schippers, C.J. Hamon, S.,Nèble, M.-H., Vesvres, V. Barsotti, S. Poulain and C. Le Marrec. 2011. Biogeochemical Processes in a Clay Formation In-situ Experiment: Part D - Microbial Analyses - Synthesis of Results. Applied Geochemistry. In press.

Wersin, P., S. Stroes-Gascoyne, F.J. Pearson, C. Tournassat, O.X. Leupin and B. Schwyn. 2011. Biogeochemical Processes in a Clay Formation In-situ Experiment: Part G - Key interpretations & conclusions. Implications for repository safety. Applied Geochemistry. In press.

Conference Presentations & Proceedings Papers

Bea, S.A., K.U. Mayer and K.T.B. Macquarrie. 2010. Hydromechanical and Geochemical Coupling within an Intercratonic Sedimentary Basin Affected by Glaciation/deglaciation Events. Goldschmidt Conference 2010, Knoxville, Tennessee, USA. June 13-18.

Beauregard, Y., M. Gobien and F. Garisto. 2010. The dissolution and transport of radionuclides from used nuclear fuel in an underground repository. COMSOL Conference 2010, Boston, USA, Oct. 7-9.

Ben Belfadhel, M. 2010. The Long-term Management of Used Nuclear Fuel in Canada- A Geoscientific Perspective. GeoCanada 2010, Calgary, Alberta, May 10-14.

Broczkowski, M., P.G.Keech, J.J.Noel and D.W.Shoesmith. The Role of Dissolved Hydrogen in the Corrosion/Dissolution of Spent Nuclear Fuel, Electrochemical Society Meeting, Las Vegas, NV, October 10-15, 2010. Invited.

Freire-Canosa, J. 2010. Delayed Hydride Cracking Properties of the Endplate Resistance Welds of CANDU Fuel Bundles. 11th International Conference on CANDU Fuel. Niagara Falls, Canada. Oct. 17-20.

Freire-Canosa, J. 2010. Deformation of Unirradiated CANDU Fuel Elements under Bending Loads. 11th International Conference on CANDU Fuel, Niagara Falls, Oct. 17-20.

Garisto, F. 2010. Results of a safety assessment of a glaciation scenario for a used fuel repository. BIOPROTA Annual Meeting, Stockholm, Sweden, May 24-27, 2010.

He, H., Z.Qin, D.Zagidulin and D.W.Shoesmith. The Key Properties Controlling the Chemical Reactivity of Uranium Dioxide, International Spent Fuel Workshop, Barcelona, Spain, November 3-5, 2010.

He, H. Z.Qin and D.W.Shoesmith. The Influence of Non-stoichiometry on the Corrosion Kinetics of Uranium Dioxide, Electrochemical Society Meeting, Las Vegas, NV, October 10-15, 2010.

Kwong, G. 2010 Performance of a Carbon Steel Container in a Sedimentary Rock Deep Geological Repository. MS&T 2010 Conference, Houston, Texas, USA. Oct 16-20.

Perras, M.A., M. S. Diederichs and T. M. Lam. 2010. A Review of Excavation Damage Zones in Sedimentary Rocks with Emphasis on Numericxal Modelling for EDZ Definition. GEO2010, Calgary, Canada. Oct. 2010.

Schneider, G., S. Emsley and Davis, R.K. 2010. Review of Geophysical Methods to Support Evaluations of Potential Candidate Sites for a Used Nuclear Fuel Deep Geological Repository. GeoCanada 2010, Calgary, Alberta, May 10-14.

Shek, G.K., B.S. Wasiluk, J. Freire-Canosa and T. Lampman. 2010. Delayed Hydride Cracking Properties of the Endplate Resistance Welds of CANDU Fuel Bundles. 11th International Conference on CANDU Fuel. Niagara Falls, Canada. Oct. 17-20.

Sheppard, S. 2010. Assessment of Se-79 in the biosphere. BIOPROTA Annual Meeting, Stockholm, Sweden, May 24-27, 2010.

Sheppard, S. 2010. New transfer factors to fish, wild game, domestic animals and crops. BIOPROTA Annual Meeting, Stockholm, Sweden, May 24-27, 2010.

Invited Presentations

Hirschorn, S. and E. Mantagaris. 2010. Long-Term Management of Canada's Used Nuclear Fuel. Shad Valley Presentation, Queen's University, July 19.

Hirschorn, S. and E. Mantagaris. 2010. Long-Term Management of Canada's Used Nuclear Fuel. Shad Valley Presentation, Lakehead University, July 9.

Hirschorn, S. and E. Mantagaris. 2010. Long-Term Management of Canada's Used Nuclear Fuel" Shad Valley Presentation, University of New Brunswick, July 15.

Kwong, G, 2010. Cost of Used Fuel Retrieval. OECD NEA Radioactive Waste Management Committee, R&R Working Group, Paris, June 22.

Kwong, G, 2010. Long-term management of Canada's used nuclear fuel. University of Ontario Institute of Technology (UOIT), November 1.

Murchison, A. and J. McKelvie. 2010. Long-term management of Canada's used nuclear fuel. Ryerson University, October 8.

A.3 Scholarships

NWMO awarded the following students industrial postgraduate scholarships in collaboration with the Natural Sciences and Engineering Research Council of Canada:

Andres, Heather. Anthropogenic Forcing of the Greenland Ice Sheet Mass Balance: Regional Climate Responses and Feedbacks. University of Toronto. Supervisor Dr. Dick Peltier.

Ghazvinian, Ehsan. Fracture initiation and propagation in sedimentary rocks: Implications for excavation damage zone (EDZ). Queen's University. Supervisor Dr. Mark Diederichs.

Henkemans, Emily. Interaction between a continental ice sheet and groundwater, Kangerlussuaq, West Greenland. University of Waterloo. Supervisor Dr. Shaun Frape.

Makahnouk, Mike. Water/Rock Interaction Related to Mineralogy, Paleoclimate, and Long Term Rock Stability Studies. University of Waterloo. Supervisor Dr. Shaun Frape.

Perras, Matthew. Investigation of the Development and Behaviour of Excavation Damage Zones Associated with Tunnel Construction for Nuclear Waste Repositories in Sedimentary Rocks: Applications for Optimization of Excavation Method and EDZ Cut-off Design. Queen's University. Supervisor Dr. Mark Diederichs.

Saso, Joe. Hydrogeochemical investigation of diagenesis and fluid-migration history in sedimentary basins. University of New Brunswick. Supervisor Dr. Tom Al.

NWMO co-funded a MITACS Accelerate internship to:

Beauregard, Y. (Master's). Corrosion behaviour of UO_2 and modelling radionuclide releases from defective containers. University of Western Ontario, Chemistry Department. Supervisors Dr. D. Shoesmith and Dr. S. Rohani. NWMO Supervisor: F. Garisto.

A.4 LIST OF RESEARCH COMPANIES, CONSULTANTS AND UNIVERSITIES

AECOM
Alberta Innovates – Technology Futures
AMEC-NSS
Amphos21
Atomic Energy of Canada Ltd.
BJ Machine
Candesco Corporation
Chandler, N.
Columbia University (Dr. L. Seeber and Dr. K. Jacob)
David P. Jackson & Associates Ltd.
ECOMatters Inc.
Engineering Simulations Inc.
FSS Canada Consultants Inc.
Gascoyne GeoProjects Inc.
Geofirma Engineering
Golder Associates Ltd.
G.R. Simmons & Associates Consulting Services Ltd.
HydroGeoLogic Inc.
Integrity Corrosion Consulting Ltd.
John Sims and Associates
Kinectrics Inc.
March Consulting Associates Inc.
McGill University (Dr. P. Selvadurai)
Natural Resources Canada (formerly Geological Survey of Canada)
New Mexico Institute of Mining and Technology (Dr. M. Person)
Nuclear Safety Solutions Limited
Queen’s University (Dr. M. Diederichs)
Quintessa Inc.
Royal Military College of Canada (Dr. G. Siemens, Dr. B. Lewis)
RSRead Consulting Inc.
SENES Consultants Ltd.
Serco Technical and Assurance Services
SKB International Consultants
SNC Lavalin
Université Laval (Dr. R. Therrien)
University of Bern (Dr. M. Mazurek)
University of British Columbia (Dr. U. Mayer)
University of Calgary (Dr. P. Wu)
University of Manitoba (Dr. J. Blatz)
University of New Brunswick (Dr. T. Al, Dr. K. MacQuarrie)
University of Ontario Institute of Technology (Dr. B. Ikeda)
University of Ottawa (Dr. I. Clark)
University of Toronto (Dr. R.C. Newman, Dr. P. Young, Dr. W. Peltier, Dr. B. Sherwood Lollar)
University of Waterloo (Dr. S. Frape, Dr. E. Sudicky, Dr. J. Sykes, Dr. S. Normani, Dr. Y. Yin)
University of Western Ontario (Dr. D. Shoesmith, Dr. G. Atkinson)
U.S. Geological Survey

APPENDIX B: ABSTRACTS FOR TECHNICAL REPORTS FOR 2010

ABSTRACT

Title: **Technical Program for Long-Term Management of Canada's Used Nuclear Fuel – Annual Report 2009**
Report No.: **NWMO TR-2010-01**
Author(s): J. McKelvie, M. Ben Belfadhel, K. Birch, J. Freire-Canosa, M. Garamszeghy, F. Garisto, P. Gierszewski, M. Gobien, S. Hirschorn, N. Hunt, A. Khan, E. Kremer, G. Kwong, T. Lam, H, Leung, P. Maak, C. Medri, A. Murchison, S. Russell, M. Sanchez-Rico Castejon, U. Stahmer, E. Sykes, A. Urrutia-Bustos, J. Villagran, A. Vorauer, T. Wanne and T. Yang
Company: Nuclear Waste Management Organization
Date: March 2010

Abstract
This report is a summary of progress in 2009 for the Nuclear Waste Management Organization's (NWMO's) Technical Program. The Technical Program is supporting implementation of Adaptive Phased Management (APM), Canada's approach for long-term management of used nuclear fuel.

Significant technical program achievements in 2009 include:

- NWMO established an arrangement with the Canadian Nuclear Safety Commission to review and assess NWMO conceptual designs and safety for APM during the pre-licensing phase.
- The NWMO Independent Technical Review Group (ITRG) held their second review of the NWMO technical program. The ITRG noted significant developments in the work since 2008, and indicated that the program covers a full range of scientific and technical topics that are relevant to the current stage of APM. NWMO prepared an action plan addressing the recommendations of the ITRG report.
- NWMO continued to participate in international research activities associated with the SKB Äspö Hard Rock Laboratory, Mont Terri Rock Laboratory, Greenland Analogue Project, Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency Research Projects and the international working group on biosphere modelling (BIOPROTA).
- NWMO provided research contracts and research grants to 13 Canadian universities and, as an approved industrial partner with the Natural Science and Engineering Research Council of Canada, awarded 3 scholarships for Ph.D. students in 2009.
- The NWMO technical program expanded in 2009, with the addition of 7 new staff. NWMO's research program published 28 NWMO technical reports and 9 peer-reviewed journal articles.
- NWMO conducted research on: used fuel integrity; used fuel container corrosion; sealing material properties and their behaviour in the subsurface; and biosphere transfer processes. NWMO also continued to develop a repository monitoring and retrieval program and continued to survey developments in reprocessing and alternative waste management technologies.
- NWMO initiated projects to refine engineering conceptual designs, cost estimates, transportation logistics and implementation schedules in support of APM. These projects will be completed in the 2010/2011 time period.
- NWMO developed a set of technical site evaluation criteria and a stepwise site evaluation process to support the draft Site Selection Process Proposal for public review.

- The NWMO geoscience program continued to develop and assess laboratory and field characterization tools and methods in support of the site selection process. These included methods pertaining to: geophysical investigation; assessment of seismicity; matrix porewater characterization; assessment of radionuclide transport processes; fracture network modelling; and assessment of excavation damage zones.
- NWMO also continued to develop numerical modelling methods and continued to assess long-term geosphere stability associated with glaciations, seismicity and deep groundwater systems.
- NWMO continued to maintain and improve models and data suitable for supporting the safety assessment of potential sites and designs. In 2009, models assessing the waste form, repository, geosphere and biosphere were improved and maintained.
- A glaciation scenario safety assessment was completed for a deep geological repository for used nuclear fuel. Preparatory work was conducted in 2009 for the “Fourth Case Study” which will examine the updated used fuel repository design (with in-floor placement of larger containers) at a hypothetical site in crystalline rock.
- NWMO staff gave invited lectures at several Canadian Universities and organized and chaired sessions at the Spent Fuel Workshop (Toronto, 7-8 May, 2009) and the American Geophysical Union Joint Assembly (Toronto, 24–27 May, 2009).

ABSTRACT

Title: **Groundwater equilibration and radionuclide solubility calculations**
Report No.: **NWMO TR-2010-02**
Author(s): Lara Duro, Vanessa Montoya, Elisenda Colàs and David García
Company: Amphos 21
Date: December 2010

Abstract
In a geological repository, containers are designed to prevent groundwater from getting in contact with fuel. In the case of a defective container water may contact the fuel and hence radionuclides might be mobilized. The assessment of radionuclide solubility limits in the near field of a repository for used fuel is essential for the safety demonstration of such as a disposal system. The objective of this report is then the study of the solubility of 19 radionuclides (Am, As, Bi, C, Cu, Mo, Nb, Np, Pa, Pb, Pd, Pu, Ra, Se, Sn, Tc, Th, U and Zr) in two different groundwater compositions, considered as representative groundwaters expected in the geological formation likely to host a future Canadian geological repository: the saline crystalline groundwater (CR-10) and the highly saline sedimentary groundwater (SR-270) provided by NWMO.

The crystalline groundwater CR-10 is well known in previous works performed by NWMO which has undergone thermodynamic equilibration calculations with minerals present at repository depth and with the near field components (bentonite buffer and the container). The sedimentary groundwater SR-270 is not as well established. This report summarise the calculation of the groundwater composition resulting from the interaction of SR-270 sedimentary groundwater with the MX-80 bentonite buffer and the carbon steel insert (SA 516 grade 70) of the Cu-Container.

Solubility calculations have been carried out with groundwaters equilibrated with minerals present at repository depth (*CR-10 eq* and *SR-270 eq*) and groundwaters equilibrated with the bentonite buffer and the container (*CR-10 NF* and *SR-270 NF*).

The Yucca Mountain Pitzer database (data0.ypf.R2) has been used in the solubility assessment of Am, C, Cu, Np, Mo, Pu, Tc, Th, U and Zr. The ThermoChimie v.7b SIT (Specific Ion Interaction Theory) database (released as sit.txt with the code PHREEQC v.17) has been used to compare the results obtained with the Yucca Mountain Pitzer database. The SIT database has been used for the solubility assessment of the elements not included in the Yucca Mountain Pitzer database.

The ThermoChimie/SIT database does not include thermodynamic data for the element Bi and the thermodynamic data for elements As and Cu are not complete. A literature review of the available information has been carried out to find a consistent and alternative thermodynamic dataset for Bi, As and Cu solubilities.

Sensitivity analyses, studying the influence of some key parameters on the solubility of radionuclides, and qualitative uncertainty analysis, describing the conceptual uncertainties that may affect the solubility limits of a radionuclide, are also provided.

ABSTRACT

Title: **Investigations of Diffusive Transport Processes in Sedimentary Rock**
Report No.: **NWMO TR-2010-04**
Author(s): Lisa Cavé, Tom Al, Yan Xiang and Diana Loomer
Company: University of New Brunswick
Date: June 2010

Abstract
Sedimentary rocks of low permeability and porosity are under investigation in Canada as potential host rocks for a deep geological repository for the long term containment of used nuclear fuel. One of the benefits of these rock types for this purpose is that they minimize, and in some cases may eliminate, advection. This results in extremely slow, diffusion-dominated transport of solutes in ground water. The overall objective of the research reported here was to develop methodologies and improve the knowledge of mechanisms controlling solute transport in diffusion-dominated sedimentary rock systems. This project focused on the quantification of diffusion and reaction processes, and rock pore-water characterization.

A study of rock diffusion properties across a sequence of Ordovician shale was conducted to investigate the scale-dependency of diffusive transport and the variations in diffusion properties with changes in lithology. Diffusion coefficients and diffusion-accessible porosity were measured on centimetre-scale samples from shale layers, and limestone and siltstone interbeds (hard beds), from across the Georgian Bay Formation, in the Michigan Basin. Diffusion measurements were made by radiographic and through-diffusion techniques using conservative aqueous tracers. Porosities and diffusion coefficients were found to be strongly dependent on the rock type, with higher porosities and diffusion coefficients measured for the shale than the hard beds. This information was combined with a very detailed geological log to determine the directional anisotropy in diffusion properties at the formation scale, identify a trend of slightly increasing diffusion coefficients with depth across the Georgian Bay Formation, and to develop a protocol for characterizing the variability in diffusive properties across a rock formation using the relationships between diffusion and physical and mineralogical properties of the rock.

A new radiography technique was also developed to quantify diffusion-reaction processes using a non-conservative tracer, cesium. The technique is based on the measurement of tracer concentration profiles as a function of time and was tested on samples of Queenston Formation shale. Reactive-transport modelling techniques were used to match experimental data and estimate mean cesium diffusion coefficients ($D_{p\ Cs} = 7.5 \times 10^{-11}\ m^2/s$), cesium exchange selectivity coefficients ($\log k_{Cs+/Na+} = 1.1$) and cation exchange capacity (CEC = 8.3 meq/100 g) for the intact samples. The estimated values for cesium are all slightly lower than measurements made by other techniques on argillaceous materials under consideration by national organizations for long-term radioactive waste management in Europe.

ABSTRACT

Title:	**The Effect of $CaCl_2$ Porewater Salinity (50-100 g/L) on the Culturability of Heterotrophic Aerobic Bacteria in Compacted 100% Bentonite with Dry Densities of 0.8 and 1.3 g/cm^3**
Report No.:	**NWMO TR-2010-06**
Author(s):	S. Stroes-Gascoyne, C.J. Hamon, D.A. Dixon and D.G. Priyanto
Company:	Atomic Energy of Canada Limited
Date:	April 2010

Abstract
This report presents and compares the results of a study on the culturability of indigenous microbes in compacted bentonite infused with calcium chloride ($CaCl_2$) porewater, with earlier data obtained with sodium chloride (NaCl) porewater. This study was carried out to assess whether, in saline Ca-dominated groundwaters, the salinity effects on microbes indigenous to Wyoming MX-80 bentonite would be similar to those determined with NaCl solutions in previous work. The results from this study show that aerobic culturability results obtained for porewaters with $CaCl_2$ are largely similar to the NaCl porewater results. Low aerobic culturability was observed for salt concentrations of ≥ 50 g $CaCl_2$/L. The lower culturability at a salt concentration of 50 g $CaCl_2$/L compared to 50 g NaCl/L would need to be confirmed.

ABSTRACT

Title: **Anaerobic Corrosion Studies of Carbon Steel Used Fuel Containers**
Report No.: **NWMO TR-2010-07**
Author(s): Roger C. Newman[1], Steve Wang[1], and Gloria Kwong[2]
Company: [1]University of Toronto, [2]Nuclear Waste Management Organization (NWMO)
Date: September 2010

Abstract

The Canadian nuclear waste management concept envisages using carbon steel containers as one of the container design options for containing and isolating used nuclear fuel waste in a deep geological repository. Steel corrosion in anticipated repository environments has been studied, but was mostly focused in two main areas: (i) aerobic or oxygen containing environments (both in vapour and liquid phases); and (ii) anaerobic, solution environments. The atmospheric corrosion behaviour of steel in a humid, anaerobic or anoxic environment, is a new topic, with virtually no published data to rely on.

A program of experimental work was undertaken to improve the existing knowledge of anaerobic, atmospheric corrosion of carbon steel. Atmospheric corrosion testing was conducted on carbon steel wires in anoxic atmospheres at 30, 50 and 70^{o}C, over a wide range of relative humidity (30-100% RH), with and without sodium chloride (NaCl) contamination of the wire surfaces. Hydrogen evolved from corrosion was initially monitored with a high sensitivity pressure gauge system and converted to estimated corrosion rate. The sensitivity of such measurements is ca. 0.005 $\mu m \cdot y^{-1}$. With large amounts of salt contamination, sustained final corrosion rates in the range of 0.01 to 0.8 $\mu m \cdot y^{-1}$ were observed over test durations of 935 to 1725 hours (tests of such length were not undertaken at 70^{o}C, but higher corrosion rates may be expected). It appears that corrosion occurred at RH values below that associated with saturated NaCl solution. Without salt contamination, the corrosion rates are very low, and can only be detected using a solid-state electrochemical hydrogen sensor. The hydrogen sensor can detect the pressure increase down to a limit of ca. 0.1 Pa, corresponding (depending on the exact procedure) to a corrosion rate as low as ca. 0.0001 $\mu m \cdot y^{-1}$. The estimated corrosion rates for the degreased and pickled wires were found to be < 0.01 $\mu m \cdot y^{-1}$. The use of a quartz crystal microbalance as an alternative way to study such phenomena was found to be unpromising, owing to the lack of long-term stability of the commercial equipment evaluated.

In parallel with the corrosion experiments, corrosion product surface analyses were performed. The oxides formed on the carbon steel surface were examined using (i) scanning electron microscopy coupled with energy dispersive X-ray analysis (SEM/EDX) to determine the structure of the corrosion product films, (ii) X-ray photoelectron spectroscopy (XPS) to identify the chemical composition of the films, and (iii) Raman and Fourier Transform Infrared spectroscopy (FTIR) to study bonding. Oxides formed on steel surfaces were found to consist mostly of Fe_3O_4, with some Fe III species from traces of air exposure; carbonate was detected on the NaCl contaminated surfaces which had been subject to a degree of prior aerobic corrosion. A high humidity (100%) environment produces more loose surface oxide than a lower humidity environment (75%). The experimental results of this study will be applied to assess the corrosion behaviour of carbon steel containers during the anoxic, unsaturated phase in a deep geological repository.

ABSTRACT

Title: **The Effects of Elevated Temperatures on the Viability and Culturability of Bacteria Indigenous to Wyoming MX-80 Bentonite**
Report No.: **NWMO TR-2010-08**
Author(s): S. Stroes-Gascoyne and C.J. Hamon
Company: Atomic Energy of Canada Limited
Date: April 2010

Abstract
This report describes the results from a two-part study. In the first part, the amount of viable-but-not-culturable (VBNC) cells occurring in saturated compacted Wyoming MX-80 bentonite with a dry density range of 0.8 – 2.0 g/cm^3 (as determined from Phospholipid Fatty Acid (PLFA) measurements), was compared to the amount of culturable cells. It was also investigated if, upon deliberate induction of complete cell death in such bentonite samples (by sterilization at 121^{o}C) the PLFA content decreased accordingly. Results showed large differences between the numbers of VBNC cells and culturable cells in compacted bentonite, especially in samples with high dry density, suggesting that many more viable (but likely inactive) cells than culturable (possibly active) cells were present in these samples. Results also suggested that either the hydrolysis process that degrades PLFA is slow or that biological (enzymatic) activity is needed for PLFA to degrade upon cell death. In case of the latter, PLFA from dead cells may be preserved in environments with low biological activity, such as in highly compacted bentonite with low a_w values and possibly in sedimentary clay-rich environments.

In the second part, the effects of moderately elevated temperature and desiccation (60^{o}C) on the culturability of microorganisms in compacted bentonite were studied, including subsequent recovery of culturability at room temperature. In addition, the effects of higher temperatures (80^{o}C-130^{o}C) on the culturability of microbes in compacted bentonite plugs were investigated. Results showed that the few culturable cells in highly compacted bentonite plugs were not particularly sensitive to a temperature of 60^{o}C and concurrent desiccation. However, the large number of culturable cells in lower dry density bentonite plugs was reduced by up to five orders of magnitude at 60^{o}C. Some culturability remained after exposure to 80^{o}C at all dry densities. At 121^{o}C and 130^{o}C all culturability was below the detection limit for low dry density samples. A very low level of (anaerobic) culturability was observed at high dry density, even after exposure to 130^{o}C. The difference in sensitivity to temperature is thought to be due to the difference in the physiological state of the cells present in the samples (e.g., vegetative cells or spores).The results also showed that the large effects of temperature on culturability in low dry density bentonite were reversible when the heat source was removed and re-saturation was allowed to occur.

The results from the PLFA and temperature studies collectively suggest that microbial cells may remain viable in highly compacted bentonite despite high temperatures, high swelling pressures, desiccation and low water activity. The presence of viable cells implies the potential for increased microbial activity under more favourable conditions in bentonite, such as lower dry density. It is, therefore, important that a high dry density is maintained throughout the bentonite in a repository to keep microbial activity to a minimum. Compliance models can be used to determine the required as-placed dry density of bentonite buffer and gap fillings to achieve specific targets for long-term equilibrium dry densities for various placement room designs.

ABSTRACT

Title: **Glaciation Scenario: Groundwater and Radionuclide Transport Studies**
Report No.: **NWMO TR-2010-09**
Author(s): Robert Walsh and John Avis
Company: Intera Engineering
Date: July 2010

Abstract

Previous Canadian safety assessments have focussed on radionuclide transport from a deep geologic repository under steady climate and groundwater conditions The purpose of the current study was to address potential impacts associated with glacial cycles.

The effects of glaciation were addressed using a three-dimensional (3-D) numerical model of a hypothetical repository in the Canadian Shield. This study investigates: (1) the impact on the hydrogeological system of a representative glacial cycle, consisting of temperate (present day climate), permafrost, ice cover and proglacial lake periods, and, (2) the impact of multiple glacial cycles on the long-term transport of radionuclides from a hypothetical defective container. Additional numeric studies were performed to assess the depth of glacial meltwater intrusion. The model included a representative set of fractures across the 13 km x 20 km model domain. Salinity effects were not included.

The current study shows that the location of the shortest groundwater pathway from the repository to surface may be different for the transient glaciation flow model. Factors such as the proximity of fracture zones are common to both steady state and transient models, but other important factors determining travel time, such as proximity to taliks, will be unique to the transient model.

Transient groundwater flow results show that hydrogeological conditions are profoundly changed during climate cycles, with median groundwater velocities within the repository footprint varying over two orders of magnitude. Glaciation induced pressure gradients are rapidly propagated horizontally and vertically through the highly connected high permeability fracture system and through hydromechanical coupling using a simplified one-dimensional strain hydromechanical model.

Although the glacial cycles do modify the flow field extensively, the cumulative impact on radionuclide transport of repeated cycles of advance and retreat tend to effectively “cancel out", leading to a general plume structure that is not substantially different from the steady-state flow and transport model. It was also seen that glacially induced overpressures can persist for thousands of years after the glacier itself has retreated, and this effect impacts transport to the surface. The study also showed that taliks (zones where permafrost is absent underneath surface water features) can have a very significant role in transmitting glacial recharge into the deep geosphere, and in acting as discrete hydraulic windows through which overpressurized water from preceding glaciation events is drained.

The Reference Case model predicted higher ^{129}I concentrations in well water (albeit for limited time periods immediately after ice retreat), higher peak ^{129}I mass flows to the biosphere, and higher average ^{129}I mass flows to the biosphere (as indicated by higher cumulative mass flow) compared to the Steady State Temperate climate model. Peak well concentrations and mass flows in the glaciation model also occurred earlier than in the steady state model. The steady state model was run with and without a well. Mass flow comparisons were done against the model without a well, as the well intercepted the bulk of the ^{129}I that would otherwise reach the small lake/talik above the repository.

Several uncertainties were considered such as the appropriate boundary conditions, higher storage coefficients, and the presence or absence of taliks under ice sheets. The broad conclusions were not very sensitive to these changes. The modelling project did not incorporate other potential sources of variation, such as alternative permeability values. However, the guiding philosophy was to choose values and assumptions that, while remaining within the bounds of hydrogeological realism, would overpredict radionuclide transport when a more detailed sensitivity analysis was not possible.

ABSTRACT

Title: **Glaciation Scenario: Safety Assessment for a Deep Geological Repository for Used Fuel**
Report No.: **NWMO TR-2010-10**
Author(s): F. Garisto[1], J. Avis[3], T. Chshyolkova[2], P. Gierszewski[1], M. Gobien[1], C. Kitson[2], T. Melnyk[2], J. Miller[2], R. Walsh[3] and L. Wojciechowski[2]
Company: [1]Nuclear Waste Management Organization, [2]Atomic Energy of Canada Limited and [3]Intera Engineering
Date: July 2010

Abstract
A key part of the long-term management of used fuel in Canada is the containment and isolation of the used fuel in a deep geological repository. The reference time frame for the safety assessment of such repositories is one million years, roughly equivalent to the time scale for the radioactivity in the used fuel to decrease to that due to its natural uranium content. Over the past one million years, the most significant natural event across Canada has been repeated glacial cycles. In principle, these will continue since the long-term variation in solar insolation will continue.

In previous Canadian safety assessments of a used fuel repository, the effects of glaciation were considered in geoscience studies and in the engineering design of the repository. However, the potential impacts of glaciation on the safety and performance of the repository were only qualitatively evaluated.

In the current safety assessment study, the effects of an evolving climate with multiple glaciations are quantitatively evaluated. This study is referred to as the Glaciation Scenario to distinguish it from the constant climate scenarios, which presume an unvarying climate, previously investigated. For this study, the repository is assumed to be located at the hypothetical site on the Canadian Shield used in the Third Case Study.

The detailed three-dimensional modelling results confirm the expected significant impact of glaciation on the groundwater flow system, affecting velocity direction and magnitude. The effects are greatest near the surface but extended to the repository level. The open taliks during permafrost periods, in particular, are a dominant factor, focusing system impacts at a discrete location. (A talik is a layer of unfrozen ground that lies in permafrost areas.) Furthermore, the transport calculations indicate that radionuclide mass flows to the surface biosphere are quite different for the transient glaciation model compared to the equivalent constant climate case, with mass flows in the glaciation model both larger and smaller than in the constant climate case. Nevertheless, the overall trends and cumulative mass flows to the biosphere are similar in these two very different climate cases.

For the Glaciation Scenario, the safety assessment calculations indicate that the calculated dose rates are highest during temperate periods. This occurs because only the critical group living during temperate periods uses a well, rather than a lake, as its domestic water source and radionuclide concentrations in well water are typically several orders of magnitude higher than in lake water, leading to higher exposures. In the Reference Case of the Glaciation Scenario, the calculated peak total dose rate is about 3.7×10^{-7} Sv/a, with I-129 contributing the most to the total dose rate. This is similar to the peak dose rate of 1.3×10^{-7} Sv/a for the corresponding case in constant climate scenario. Both these dose rates are well below the dose rate constraint of 3×10^{-4} Sv/a recommended by International Commission on Radiological Protection (ICRP) for disposal of long-lived solid radioactive waste and the average Canadian natural background dose rate of 1.8×10^{-3} Sv/a.

A series of sensitivity cases and probabilistic cases were also investigated. As would be expected, the "what if" sensitivity case in which all containers fail simultaneously gives the highest calculated dose rate. Even for this improbable sensitivity case, the dose rate exceeds the ICRP dose rate constraint of $3x10^{-4}$ Sv/a for only a brief period of time, at the end of several glacial cycles, but remains well below the average Canadian background dose rate of $1.8x10^{-3}$ Sv/a.

The results of the Climate State Duration probabilistic case suggests that varying the glacial cycle could lead to higher calculated dose rates compared to the Reference Case of the Glaciation Scenario. Although the 90th percentile calculated dose rate in this probabilistic simulation was 3-fold higher than in the Reference Case, it remained well below the dose rate constraint recommended by ICRP.

In summary, for the hypothetical site and repository of the Third Case Study, calculated peak dose rates for the Glaciation Scenario are approximately of the same order of magnitude as for the corresponding constant (temperate) climate scenario. The calculated peak dose rates for the Glaciation Scenario are well below the ICRP dose constraint and the average natural Canadian background dose rate. Thus, it can be concluded that for the hypothetical Third Case Study site and repository, the impacts of a deep geological repository would be well below regulatory limits when the effects of glaciation are considered.

ABSTRACT

Title: **Initial Evaluation of Mechanical Stress Distributions in Spent CANDU Fuel Bundles**
Report No.: **NWMO TR-2010-11**
Author(s): Timothy Lampman and Adrian Popescu
Company: AMEC NSS Limited
Date: June 2010

Abstract
This report describes an initial analysis of the mechanical stress distributions in irradiated 28-element fuel bundles at the start of interim dry storage. The finite element CANDU fuel bundle models previously developed using unirradiated material properties have been updated to include irradiation effects, such as material properties to account for fast-neutron irradiation and consideration of changes to the original bundle geometry following irradiation. Calculations of the stress distribution were performed for different post-discharge bundle geometries modelled under initial dry storage conditions. From the stress distributions within the CANDU fuel bundles, stress intensity factors for the weld notch of the endplate-to-endcap welds were performed for future comparison with experimentally-determined critical stress intensity factors required for delayed hydride cracking.

There are uncertainties related to modelling irradiated fuel elements and where these are considered potentially significant, a conservative approach was used to bound the expected stress levels and intensity factors. The analysis indicates that the endplate-to-endcap welds act as stress concentrators and at the endplate-to-endcap welds, the calculated stress intensity factors did not exceed approximately 3 MPa $m^{1/2}$.

ABSTRACT

Title: **CANDU Fuel Element Model Development and Sensitivity Study**
Report No.: **NWMO TR-2010-12**
Author(s): Adrian Popescu and Timothy Lampman
Company: AMEC NSS Limited
Date: June 2010

Abstract
This report documents continued work on the Bundle Stress Model for CANDU fuel. New modified models with improved pellet-to-sheath and pellet-to-pellet interactions were developed to evaluate stress fields present in spent fuel during dry storage and to calculate the stress intensity factors at the endcap-to-endplate welds.

The models were compared against a new series of validation experiments using unirradiated fuel elements. Comparison of the modelled and experimental results shows a good agreement and demonstrates that the model is capable of predicting the mechanical behaviour of the 28- and 37-element fuel bundles.

Sensitivity studies confirmed the model capability to simulate bundles with dimensions and material properties within the known variability of their values.

ABSTRACT

Title: **Assessment of Delayed Hydride Cracking in Used CANDU Fuel Bundles during Dry Storage**
Report No.: **NWMO TR-2010-13**
Author(s): Timothy Lampman and Paul Gillespie
Company: AMEC NSS
Date: December 2010

Abstract
This report summarizes various experimental and modelling activities performed to evaluate if Delayed Hydride Cracking (DHC) could be occurring in used CANDU fuel stored in Ontario Power Generation Dry Storage Containers. The conclusion of this work is that DHC is not expected in the used CANDU fuel.

DHC requires a hydrogen concentration in a material that exceeds its solubility, a stress gradient that concentrates the dissolved hydrogen thus forming solid hydrides, and a sufficient stress to fracture the formed hydrides. If all conditions are present, DHC is expected to occur. Early studies identified that the hydrogen concentration in used fuel was sufficient for DHC, but it was uncertain whether the magnitude of the stress in CANDU fuel would be sufficient for DHC under dry storage conditions.

To investigate this further, AMEC NSS developed finite element stress models of CANDU fuel bundles used by Ontario Power Generation and Bruce Power. The stress models were developed so they could account for the condition of used CANDU fuel. A key area of the fuel bundle that is a concern for DHC is the endplate-to-endcap weld where there is a very sharp weld discontinuity. A case study of the fuel in dry storage conditions was performed to determine the stress intensity factors at the tip of the weld discontinuity. The results suggest stress intensity factors no larger than 3 MPa $m^{1/2}$.

In understanding if this stress intensity factor is sufficient to initiate DHC, Kinectrics developed a methodology and apparatus to determine the critical stress intensity factor for initiation of DHC at the endplate-to-endcap welds. Both 28- and 37-element fuel designs from different manufacturers were tested. The results of the testing suggest the critical stress intensity factor is in the range of 7.6 MPa $m^{1/2}$ to 13.6 MPa $m^{1/2}$.

Since the calculated stress intensity factor is well below the critical stress intensity factor, DHC is not expected to occur during dry storage. Though there are some uncertainties remaining in this work, they are not considered to be significant enough to affect the conclusion that DHC is not expected during dry storage, given the indicated margin between the applied and critical stress intensity factors.

ABSTRACT

Title: **Seismic Activity in the Northern Ontario Portion of the Canadian Shield - Annual Progress Report for the Period January 01 – December 31, 2009**
Report No.: **NWMO TR-2010-15**
Author(s): S.J. Hayek, J.A. Drysdale, J. Adams, V. Peci, S. Halchuk and P. Street
Company: Canadian Hazards Information Service, Natural Resources Canada
Date: December 2010

Abstract
The Canadian Hazards Information Service (CHIS), a part of Natural Resources Canada (NRC) continues to conduct a seismic monitoring program in the northern Ontario and eastern Manitoba portions of the Canadian Shield. This program has been ongoing since 1982 and is currently supported by a number of organizations, including the NWMO. A key objective of the monitoring program is to observe and document earthquake activity in the Ontario portion of the Canadian Shield. This report summarizes earthquake activity for the year 2009.

CHIS maintains a network of eighteen seismograph stations to monitor low levels of background seismicity in the northern Ontario and eastern Manitoba portions of the Canadian Shield. Core stations are located at: Sioux Lookout (SOLO), Thunder Bay (TBO), Geraldton (GTO), Kapuskasing (KAPO), Eldee (EEO), and Chalk River (CRLO). These are augmented by the CHIS network of temporary stations at: Sutton Inlier (SILO), McAlpine Lake (MALO), Kirkland Lake (KILO), Sudbury (SUNO), Atikokan (ATKO), Experimental Lake (EPLO), Pickle Lake (PKLO), Pukaskwa National Park (PNPO), Aroland (NANO), and Timmins (TIMO). The digital data from a temporary station at Victor Mine (VIMO), supported by the diamond mine industry, and a station at Pinawa (ULM), which has funding from the Comprehensive Nuclear Test Ban Treaty Organization (CTBTO) are also used in this study.

All the stations are operated by CHIS and transmit digital data in real-time via satellite to a central acquisition hub in Ottawa. CHIS-staff in Ottawa integrate the data from these stations with those of the Canadian National Seismograph Network and provide monthly reports of the seismic activity in northern Ontario.

During 2009, 82 earthquakes were located. Their magnitude ranged from 0.5 m_N to 3.4 m_N. The largest events included a m_N 3.4 in Kirkland Lake, ON and a m_N 2.9 in James Bay. The most westerly event in the area being studied was a m_N 1.3, located 54 km south of Kenora, ON. The 82 events located in 2009 compares with 114 events in 2008, 68 events in 2007, 83 events in 2006 and 103 events in 2005.

ABSTRACT

Title: **Laboratory Bentonite Erosion Experiments in a Synthetic and a Natural Fracture**
Report No.: **NWMO TR-2010-16**
Author(s): Peter Vilks and Neil H. Miller
Company: Atomic Energy of Canada Ltd.
Date: July 2010

Abstract
A scenario being addressed as part of the long-term performance assessment for a Deep Geological Repository is the penetration of dilute glacial meltwater to depth followed by the possible erosion of bentonite from repository sealing systems and subsequent transport of colloids to the geosphere. Such a scenario also includes the presence of an adjacent, water-bearing fracture into which the bentonite can swell, forming a gel with an expanded structure. If this gel is able to release mobile bentonite colloids at a significant rate, the buffer may lose substantial mass and sorbed contaminants may be transported into the geosphere. A laboratory experimental program was initiated to address these issues. The initial experimental phase consisted of a series of mock-up tests in which bentonite erosion was studied in a synthetic transparent fracture. Sodium and calcium rich bentonite samples, spiked with fluorescent latex tracers, were used to make round 37 mm diameter plugs, which were installed in a borehole that intersected the fracture. When water (deionized or synthetic glacial meltwater) was introduced into the fracture, the expansion of the clay into the fracture was monitored under stagnant conditions and under the influence of two different flow rates. Colloid generation and transport was studied by monitoring the concentrations of bentonite and latex colloids eluted from the fracture. These tests investigated the effects of water chemistry, clay composition, fracture aperture, flow rate and fracture slope. In the second experimental phase a longer term bentonite erosion test was performed in a natural fracture within a Quarried Block to build on the results of the previous tests and to investigate the role of aperture variability on erosion and transport. This phase provided a link between a simple laboratory system and a field-scale natural system.

The results of the mock-up and Quarried Block tests found that bentonite erosion and colloid generation were affected by water composition, bentonite composition, fracture dip and fracture aperture. Significantly higher bentonite erosion and colloid generation rates were observed with deionized water than with synthetic glacial meltwater. Transported bentonite colloids formed erosion resistant deposits that altered flow properties within the fracture. Increasing flow rate did not affect the erosion rate of bentonite when stabilized by the presence of millimolar salt concentrations in the synthetic glacial meltwater. Fracture slopes had a significant influence over bentonite colloid transport and deposition, notably under no flow conditions. Under the influence of gravity, swelling bentonite migrated to the lower part of the fracture and larger colloids became trapped leaving transport mainly to the very small, 10 nm colloids.
The results of the Quarried Block tests indicate that in waters containing millimolar amounts of dissolved salts (representative of glacial melt water) the bentonite that expands into an open fracture is likely to form stable deposits that do not readily erode and release significant concentrations of bentonite colloids.

ABSTRACT

Title: **Nuclear Fuel Waste Projections in Canada – 2010 Update**
Report No.: **NWMO TR-2010-17**
Author(s): M. Garamszeghy
Company: Nuclear Waste Management Organization
Date: December 2010

Abstract

Since the Nuclear Waste Management Organization submitted its Final Study in 2005, there have been a number of planned and proposed nuclear refurbishment and new-build initiatives which could extend the projected end of nuclear reactor operation in Canada from about 2034 to about 2085 or beyond.

The important technical features of these recent nuclear initiatives include:

- The amount of used nuclear fuel produced in Canada; and
- The type of used nuclear fuel produced in Canada;

This report updates the 2009 report [Garamszeghy, 2009] and summarizes the existing inventory of used nuclear fuel wastes in Canada as of June 30, 2010 and forecasts the potential future arisings from the existing reactor fleet as well as from proposed new-build reactors. The report focuses on power reactors, but also includes prototype, demonstration and research reactor fuel wastes held by AECL.

As of June 30, 2010, a total of approximately 2.2 million used CANDU fuel bundles (44,000 tonnes of heavy metal (t-HM)) were in storage at the reactor sites. For the existing reactor fleet, the total used fuel produced to end of life of the reactors ranges from about 2.8 to 5.1 million used CANDU fuel bundles (56,000 t-HM to 102,000 t-HM), depending upon decisions to refurbish current reactors. The lower end is based on an average of 30 calendar years of operation for each reactor (i.e. no refurbishment), while the upper end assumes that reactors are refurbished and life extended for an additional 30 calendar years of operation. The 5.1 million bundles at the upper end of the projection has been reduced from the 2009 report (5.5 million), due to OPG's decision not to refurbish the Pickering B reactors.

Used fuel produced by potential new-build reactors will depend on the type of reactor and number of units deployed. New-build plans are at various stages of development and the decisions about reactor technology and number of units have not yet been made. If all of the units which are in an advanced state of planning/public discussion or where a formal licence application has already been submitted are constructed, the total additional quantity of used fuel from these reactors could be up to 1.9 million CANDU fuel bundles (31,200 t-HM), or 21,600 PWR fuel assemblies (11,640 t-HM), or 27,000 BWR fuel assemblies (3,384 t-HM), or some combination thereof. This total is unchanged from the 2009 report.

As decisions on new nuclear build and reactor refurbishment are made by the nuclear utilities in Canada, the forecasted inventory of nuclear fuel waste will be incorporated into future updates of this report.

For NWMO preliminary planning purposes, a base case of 3.6 million bundles (which represents a point between the lower and upper end forecasts to allow for some reactors being refurbished and some not) and an alternate case of 7.2 million bundles (corresponding to maximum reactor refurbishment along with some new-build CANDU type reactors) has been adopted.

ABSTRACT

Title: **The Corrosion of Zirconium Under Deep Geological Repository Conditions**
Report No.: **NWMO TR-2010-19**
Author(s): David W. Shoesmith and Dmitrij Zagidulin
Company: The University of Western Ontario
Date: October 2010

Abstract
Zirconium alloys are widely used in nuclear reactors as fuel cladding and as reactor structural elements (i.e., CANDU reactor pressure tubes), and are therefore a component of the waste materials that could be emplaced in a deep geologic repository. For this reason, the corrosion mechanisms and rates for relevant zirconium alloys under repository conditions have been reviewed. Since titanium and zirconium alloys have many similarities, and because the data base for the corrosion of titanium alloys under repository conditions is considerably more extensive than that for zirconium alloys, the electrochemical and corrosion behaviour of both materials have been compared and evaluated. Although electrochemical studies suggest Zircaloy cladding could be susceptible to pitting, redox conditions within a failed waste container will remain reducing and unable to support this corrosion process. This leaves passive corrosion as the only corrosion mechanism. The available data indicates that the rate of passive corrosion will be very low. A conservative upper limit for the passive corrosion rate would be 20 nm/year and a reasonable value would be 5 nm/year, although some studies exist to show rates less than 1 nm/year are likely.

ABSTRACT

Title: **Evaluation of Container Placement Methods for the Conceptual Design of a Deep Geological Repository**
Report No.: **NWMO TR-2010-20**
Author(s): P. Maak[1], K. Birch[1] and G.R. Simmons[2]
Company: [1] Nuclear Waste Management Organization
[2] G.R. Simmons & Associates Consulting Services Ltd
Date: December 2010

Abstract
Three generic used fuel container placement methods for a deep geological repository have been evaluated in a qualitative manner based on technical feasibility, safety, siting, monitoring and retrieval for application in crystalline rock, hard sedimentary rock and soft sedimentary rock. The three generic used fuel container placement methods, in-floor borehole, horizontal borehole and horizontal tunnel, are derived from a review of repository concepts being developed by national radioactive waste management organizations. Each container placement method has a number of variants, which depend in part on national program designs and site specific conditions. Where these variants are significantly different, they are assessed individually.

In-floor Borehole Placement Method:
The in-floor borehole placement method has been well developed and demonstrated for a deep geological repository in crystalline rock. It is the reference container placement method for crystalline rock in Sweden and Finland and there are only minor variations in design amongst the national programs. The in-floor borehole placement method is also considered to be suitable for hard sedimentary rock, but may not be suitable for soft sedimentary rock without significant ground support and other engineering modifications.

Horizontal Borehole Placement Method:
The horizontal borehole placement method is being developed and demonstrated for a deep geological repository in crystalline rock and soft sedimentary rock. It is the alternative container placement method for crystalline rock in Sweden and Finland and it is the reference container placement method, with borehole liners, for soft sedimentary rock in France and Belgium. There are significant variations in design amongst these national programs primarily driven by the potential site conditions and the level of effort associated with monitoring and retrieval of containers. It is also considered to be suitable for hard sedimentary rock.

Horizontal Tunnel:
The horizontal tunnel placement method has been developed specifically for hard sedimentary rock and is also considered suitable for soft sedimentary rock with the addition of tunnel liners for ground support. It is the reference container placement method for sedimentary rock in Switzerland. The principal technical feasibility and safety issues relate to achieving a sufficiently high bentonite buffer as-placed dry density or ambient groundwater salinity to effectively suppress microbial activity and the potential of microbially-influenced corrosion of containers in the repository. The placement method might not be appropriate in crystalline rock since both the achievable dry density of the bentonite pellets and the groundwater salinity are likely to be too low to suppress microbial activity near the containers.

The results from this assessment suggest that all three placement methods would meet the general technical evaluation criteria within the constraints outlined above.

ABSTRACT

Title: **Stress Corrosion Cracking of Carbon Steel Used Fuel Containers in a Canadian Deep Geological Repository in Sedimentary Rock**
Report No.: **NWMO TR-2010-21**
Author(s): Fraser King
Company: Integrity Corrosion Consulting Ltd.
Date: November 2010

Abstract

Carbon steel has been proposed as a used fuel container material for a deep geological repository in sedimentary rock. The container will be subject to a number of corrosion mechanisms including stress corrosion cracking (SCC). An assessment is presented of the probability that SCC will lead to through-wall penetration of the container in the repository.

Stress corrosion cracking of carbon steel has been reported in a number of environments. The environmental conditions associated with cracking are reviewed and the controlling factors and threshold environmental and stress-related conditions identified. Cracking of pipeline steels due to the presence of either dilute or concentrated carbonate/bicarbonate solutions appear to be the most relevant for the SCC of containers in the repository.

Various mechanisms have been proposed to account for the SCC of carbon steel and , more generally, for various metal/environment systems. These mechanisms are reviewed, both in relation to the SCC of carbon steel and in terms of whether they will operate in the repository environment.

Based on the environmental and mechanistic evidence, the implications for SCC of carbon steel containers in the repository are considered. The evidence is reviewed in the context of the inherent susceptibility of the material, the corrosiveness of the environment, and the aggressiveness of the mechanical loading conditions. Possible mitigation strategies are also reviewed.

Overall, it is concluded that the probability of through-wall penetration of the container due to stress corrosion cracking is low.

ABSTRACT

Title: **Coupled Thermal-Mechanical Modelling of a Deep Geological Repository using the Horizontal Tunnel Placement Method in Sedimentary Rock using CODE_BRIGHT**
Report No.: **NWMO TR-2010-22**
Author(s): Ruiping Guo
Company: Atomic Energy of Canada Limited
Date: December 2010

Abstract
A series of three-dimensional thermal transient and thermal-mechanical (T-M) stress analyses was performed on a deep geological repository (DGR) for used CANDU fuel using the Horizontal Tunnel Placement (HTP) geometry. The DGR modelled in this document is assumed to be located at a depth of 500 m in limestone.

Based on the near-field modelling, the peak temperature of the container surface is 117.0˚C at 10 years after used fuel placement and the peak temperature of the tunnel surface is 69.0˚C at 50 years after used fuel placement.

A coupled near-field T-M model was conducted for the first 1,000 years after placement of the used fuel in a DGR. Excavation-induced mechanical stresses in the rock around the placement tunnel were studied.

The stability of the rock mass was evaluated using the modified Hoek and Brown empirical failure criterion. Excavation of the placement tunnel could potentially cause a damage zone with a thickness of 0.053 m near the placement tunnel roof. At 1,000 years after placement, this damage zone in the tunnel roof will extend to a depth of 0.211 m from the tunnel roof's original surface. At the same time, a layer of damaged rock with a thickness of 0.077 m could potentially develop on the tunnel wall.

Coupled T-M far-field analyses were used to determine the peak temperatures at various regions in the repository. The peak temperature in the rock is 42.7˚C at the centre of the repository after 1,200 years. The peak temperatures at the centre of the repository edge (727.5 m from repository centreline) and repository corner (1,265 m from repository centreline) are 27.6˚C and 20.3˚C, respectively, at 4,000 years after placement. These analyses determined that the maximum thermally induced uplift at the ground surface above the centre of the repository would be about 0.13 m. This degree of deformation was determined not to be sufficient to generate additional fractures in the rock near the ground surface.

ABSTRACT

Title: **Watching Brief on Reprocessing, Partitioning and Transmutation (RP&T) and Alternative Waste Management Technology – Annual Report 2010**
Report No.: **NWMO TR-2010-24**
Author(s): David Jackson and Kenneth Dormuth
Company: David P. Jackson & Associates Ltd.
Date: December 2010

Abstract

This is the 2010 Annual Report of the NWMO watching brief on Reprocessing, Partitioning and Transmutation (RP&T) and Alternative Waste Management Technologies. International developments are reviewed based on recently published documents and on the presentations at the Nuclear Energy Agency information exchange meeting on RP&T held in San Francisco in November 2010 where significant technical progress made since the 2008 meeting was reported. The US, several European countries and Japan are in the process of reviewing their RP&T programs. The former by means of a specially appointed Blue Ribbon Commission, whereas France and Japan have major decision points in 2012.

In this report the current status of RP&T is summarized in terms of seven key questions relevant to assessing the possibility of reprocessing of used CANDU fuel based on conclusions from all three Annual Reports. While there have been some progress in R&D on advanced closed fuel cycles, they remain many decades away from commercial deployment and will require a very large investment in nuclear infrastructure. In addition, closed fuel cycles will not eliminate the need for a deep geological repository. Therefore, there is no compelling reason at this time to alter from the reference APM strategy of a deep geological repository for used CANDU fuel.

The possible use of very deep boreholes for long-term management of used nuclear fuel was the subject of increased interest and more detailed investigation in 2010. However, its cost effectiveness for used CANDU fuel has not yet been determined.

ABSTRACT

Title: **Determination of the Threshold Stress Intensity factor and Velocity of Delayed Hydride cracking of Endplate welds in CANDU Fuel Bundles with Different Design and Manufacturers**
Report No.: **NWMO TR-2010-25**
Author(s): Gordon K. Shek
Company: Kinectrics Inc.
Date: April 2010

Abstract

The threshold stress intensity factor (K_{IH}) and crack velocity of DHC in the endplate welds of three unirradiated fuel bundles were determined. The three bundles included a GE 28-element bundle, a GE 37-element bundle and a CAMECO 37-element bundle. The results are compared with those obtained from previous tests on the endplate welds of two GE 37-element fuel bundles. There were no large differences in K_{IH} values and DHC velocities among the endplate welds of the three fuel bundles tested in the current program. The endplate welds of the three bundles have higher K_{IH} and lower DHCV values than the welds of the GE 37-element bundles tested previously.

ABSTRACT

Title: **Enhanced Sealing Project (ESP): Design, Construction and Instrumentation Plan**
Report No.: **APM-REP-01601-0001**
Author(s): D.A. Dixon, J.B. Martino and D.P. Onagi
Company: Atomic Energy of Canada Limited
Date: October 2009

Abstract
The Enhanced Sealing Project (ESP) project consists of instrumenting and monitoring a full-scale shaft seal designed to permanently seal the access shaft for Atomic Energy of Canada Limited's (AECL's) Underground Research Laboratory (URL).

The seal consists of a reinforced low heat high performance concrete component, supporting and restraining a 6-m-thick layer of bentonite clay-sand mixture that is compacted in place to seal the intersection of the shaft and a fracture zone, and is restrained on top by a second concrete component. These three units comprise the composite seal. A similar seal, but using pre-compacted blocks of a clay-sand mixture is installed in a smaller ventilation shaft. The ventilation shaft seal is not instrumented.

This report describes the scope of the project and the design and construction plans for the seal as well as the instruments to be installed.

ABSTRACT

Title: **Enhanced Sealing Project (ESP): Pre-Construction Grouting Around Seal Locations**
Report No.: **APM-REP-01601-0002**
Author(s): J.B. Martino, R. Kaatz
Company: Atomic Energy of Canada Limited
Date: June 2010

Abstract

The Enhanced Sealing Project (ESP) project consists of instrumenting and monitoring of a full-scale shaft seal installed to permanently seal the 5-m-diameter access shaft to Atomic Energy of Canada Limited's (AECL's) Underground Research Laboratory. In addition to the seal in the main shaft a second seal was installed in a 1.8-m-diameter raise-bored ventilation shaft. This report describes the grouting activities that took place in advance of the construction of these two seals.

The construction of both seals is reported separately as part of the ESP documentation process. The main shaft seal consists of two low-heat high-performance concrete components that act to restrain a 6-m-long, compacted in place, bentonite clay-sand blend component (40:60 by mass). These components work together to seal the intersection of the shaft with a fracture zone and restrict future interaction between the saline groundwater below the seal and the fresher water in the fracture zone and above.

Prior to the installation of the seals, grouting of the fracture zone adjacent to the locations of the main shaft and ventilation raise was undertaken. The intent of the grouting was, in the short term, to limit the amount of water inflow from the fracture zone into the seal locations and thereby facilitate placement of the seal components. Over the longer term, the grouting is also expected to limit the flow of groundwater around the shaft seals.